III. TECHNICAL ISSUES

A. Initial Conditions

Characterization of Yucca Mountain Site and Chlorine-36 Results

Introduction

The analysis of the environmental effects caused by emplacing radioactive waste in the proposed repository at Yucca Mountain requires an understanding of initial conditions of the site. Because the proposed repository at Yucca Mountain would be located in the unsaturated zone (UZ) in a sequence of volcanic tuffs, a major effort has been expended in investigations of the vadose zone. This has required the development of a suite of computer models to investigate different conditions in the UZ which must be coupled in an appropriate manner to the saturated zone (SZ) and validated, where possible, by comparing model predictions to observations and test results.

UZ Site-Scale Flow Model

The outgrowth of this need for a suite of models is a project at the Lawrence Berkeley National Laboratory (LBNL) to develop a three-dimensional conceptual model of the UZ in cooperation with the United States Geological Survey (USGS). Work on this project was initiated several years ago, and there have been a number of modifications. A detailed description of the status of results as of 1997 is given by Bodvarsson et al. (1997a). The UZ Site-Scale Flow Model is a central component of this project, and Figure III-1 illustrates the relationships between this model and the various process models that are being developed for the unsaturated as well as the saturated zones.

Bodvarsson et al. (1997b) state that the primary objectives of the UZ model development are to: (1) integrate the available data from the UZ into a single comprehensive three-dimensional model; (2) quantify the flow of moisture, heat, and gas through the UZ; (3) evaluate the effects of repository loading on moisture, gas, and heat flow within the mountain; and (4) provide Performance Assessment and Repository Design teams with a defensible and credible model of all relevant UZ processes. According to Bodvarsson et al. (1997b), the UZ model provides estimates for important parameters and processes such as, the spatial and temporal values of percolation flux at the repository horizon; the components of fracture and matrix flow in and below the repository horizon; and the probable flow paths from the repository to the water table.

The modeling studies summarized in the LBNL report (Bodvarsson et al., 1997a) are based on the extensive data available from more than 15 years of investigations at Yucca

Mountain. These data include saturation, *in situ* and core-sample water potentials, saturated conductivities and desaturation curves, core-sample bulk-property measurements, pneumatic monitoring, temperature data, air permeability test results, geochemical analyses, and perched water body testing. The Exploratory Studies Facility (ESF) information includes data on fracture mapping, the movement of key radionuclides present in the environment, hydrochemical processes, fracture coatings, and bulk properties from *in situ* and core sample measurements.

The incorporation of all these data into modeling studies has provided a comprehensive and complex UZ model that Bodvarsson et al. (1997b) state is representative of the important UZ flow processes such as moisture flow, capillary pressure effects, gas flow, convective and conductive heat transfer, evaporation and condensation, moisture and gas flow travel times, and transport of conservative and reactive species in the mountain. The model grid is based upon the best available geologic data, and captures the complex structural features which have been characterized by data obtained through nearly 60 boreholes that penetrate a significant portion of the mountain, in addition to data from the ESF and pavement, trench, and section studies. The model has been calibrated by comparing model predictions to observations of saturations, water potentials, temperatures, and pneumatic pressures in newly drilled boreholes, as well as gas flow changes due to the construction of the ESF. In the opinion of the LBNL investigators, the validation process and extensive data set have helped to develop confidence in the model's ability to simulate ambient conditions as well as perform predictive studies.

Chlorine-36 Studies

The LBNL report (Bodvarsson, 1997a) was published in June 1997. In September 1997, Fabryka-Martin et al. (1997) published a comprehensive report on the chlorine-36 (^{36}Cl) studies that have been conducted at Yucca Mountain. The objective of this work is to acquire geochemical data and information on the movement of radionuclides already present in the environment that are relevant to the development and testing of conceptual flow and transport models of the unsaturated zone. More than 600 samples have been analyzed for ^{36}Cl from deep and shallow boreholes, soil profiles, groundwater, and the ESF. According to Fabryka-Martin et al., these data have been used to establish lower bounds on infiltration rates, estimate groundwater ages, establish bounding values for hydrologic flow parameters governing fracture transport, and develop a conceptual model for the distribution of fast flow paths.

The most extensive set of ^{36}Cl data for Yucca Mountain is from the ESF. The quantities in the northern part of the ESF are highly variable and elevated above present background levels. At several locations, the measured signals are high enough that the authors consider them to be unambiguous indicators of at least a small component of bomb-pulse ^{36}Cl, implying that some fraction of the water at the ESF level arrived there during the past 50 years. In the southern part of the ESF, indications of the presence of ^{36}Cl are less variable and at levels equal to or slightly below present-day background.

Detailed characterization of the structural settings of the ^{36}Cl sample locations and of their relationships to structural features and infiltration rates has generally supported a proposed conceptual model for fast pathways at Yucca Mountain. In order to transmit bomb-pulse ^{36}Cl to the sampled depth within 50 years, the modeling assumptions require: (1) the presence of faults that cut the PTn unit and increase its fracture conductivity; (2) sufficiently high infiltration to initiate and sustain fracture flow through the PTn layer; and (3) less than 3 meters of soil cover. The model was used to predict the distribution of bomb-pulse ^{36}Cl for the study area, including the planned East-West drift. A case-by-case evaluation by Fabryka-Martin et al. (1997) demonstrated that the model successfully predicted the presence of bomb-pulse ^{36}Cl in most cases, but did not adequately account for the apparent lack of bomb-pulse ^{36}Cl in the southern part of the ESF.

Cl concentrations measured in porewater from the PTn in the North Ramp range from 15 to 45 mg/L and, based on their low Br/Cl ratios, have not been influenced by ESF construction water. These low Cl concentrations are consistent with the Flint et al. (1996) infiltration model. Their uniformity suggests that the flux through the PTn matrix is on the order of 5 mm/year at this location. Also, because the lower values approach those measured in perched water at Yucca Mountain, Fabryka-Martin et al. (1997) state that these results support a conceptual model that does not need to invoke fracture flow through the PTn to explain the perched water chemistry.

The ^{36}Cl data are consistent with ^{14}C data and, with the results that solute-transport simulations suggest that groundwater travel times are less than 10,000 years everywhere in the unsaturated zone at Yucca Mountain. Low ^{36}Cl ratios measured for some samples from the southern part of the ESF require further evaluation in order to assess whether these ratios provide evidence for longer groundwater travel times.

Implications of Environmental Tracers Studies on Results from Flow and Transport Models

Fabryka-Martin et al. (1997) state that some discrepancies exist between the ^{36}Cl data, the conceptual model for flow and transport, and the numerical solute transport simulations. They indicate that actions needed to resolve these discrepancies include a re-assessment of PTn hydrologic properties, the incorporation of porewater Cl concentrations into the flow-model calibration process, independent evidence to confirm the infiltration model, corroborating evidence to confirm the bomb-pulse ^{36}Cl results, and an expanded data base of porewater Cl measurements.

To calibrate the UZ model, Bodvarsson et al., (1997a) used data on radionuclides present in the environment including bomb-pulse ^{36}Cl data. They concluded that the bomb-pulse ^{36}Cl found in the repository horizon represents only a small fraction of the water migrating through fractures and is therefore not helpful in estimating average percolation fluxes. However, these data can be used to infer localized “fast path” water flow. They also concluded that some of the bomb-pulse ^{36}Cl may be masked by variations in the total chloride concentration used to calculate the $^{36}Cl/Cl$ ratios and therefore cannot be relied on to identify all of the fast paths.

Bodvarsson et al (1997a) also used other geochemical data in their model calibration activities including total Cl, Sr, $^{87}Sr/^{86}Sr$, and ^{14}C. As pointed out by Fabryka-Martin et al. (1997), these data are limited, but Bodvarsson et al. (1997a) state that they yield important information about fluid flow patterns, evaporative and condensation processes, rock/water interaction, percolation flux, and groundwater ages. They believe that these data are also useful in identifying fast paths and constraining flux estimates.

Conclusions

In preparing their report on the Site-Scale Unsaturated Zone Model, Bodvarsson et al. (1997a) did not have the comprehensive report on the ^{36}Cl studies (Fabryka-Martin et al., 1997) available for their review. However, the implications from the use of the environmental tracers suggest that the discrepancies mentioned above between the ^{36}Cl data and conceptual models need further attention. This is a problem of considerable complexity and one that is beyond the scope of the review assigned to the Panel. Its importance is indicated by the fact that the UZ flow model is a key process model for the Yucca Mountain Project team's strategy as it approaches the license application phase.

Proposed East-West Cross Drift in Repository Block

In March 1997, a comprehensive planning activity was undertaken to perform an Enhanced Characterization of the Repository Block (ECRB) using a new East-West cross drift. The purpose was to determine what data would most strengthen the licensing basis while complying with the limitations and constraints imposed on characterization activities. Two of the basic problems under investigation in demonstrating suitability are: (1) the collection of sufficient data to provide a reliable and defensible description of the geologic system and its behavior under present ambient as well as potential future repository conditions, and (2) the selection of a repository site that can take advantage of the best conditions for construction activities while preserving certain options in case of any unexpected developments.

To carry out the ECRB, an integrated (DOE and M&O) team was utilized to develop a plan for an exploratory drift passing through the repository block. A consolidated list of 50 criteria was developed for a crosswalk analysis. As shown in Figure III-2., several options for the location of the cross drift were considered from which a final location was selected. There were two general perspectives that influenced the cross drift configurations: one from testing and one from design/construction. Site attributes that were of interest in testing included zones of potentially higher infiltration on the western side of the block, including evidence of fast paths. A cross section through the block illustrates that the contemplated repository development would be in the middle nonlithophysal, the lower lithophysal, and the lower nonlithophysal zones of the

Topopah Spring welded tuff layer (TSw). A primary reason for testing is to examine this vertical section with respect to fracture mapping, geomechanical, and hydrologic properties.

Repository exposures to the lower nonlithophysal strata generally start in the southern part of the block. The middle nonlithophysal is seen in the East Main drift (Figure III-2) and the bulk of the repository is in the lower lithophysal. During the mapping in the existing ESF, a zone of unexpected fracturing was encountered at station 43+00. A testing perspective was that predictive modeling could be done and compared to conditions encountered in this area. Also, in the southern part of the block, the Solitario Canyon fault has a reasonable amount of displacement, and the splay on the Solitario Canyon is clearly present. A recommendation for testing was to conduct drifting along the repository alignment within the repository block near station 43+00.

The design perspective was more focused on the northern part of the ECRB, which is the preferred zone for potential expansion. The design team was concerned about an excavation in the repository horizon, because if the cross drift orientation is not coincident with the eventual repository alignment, there is a potential to lose repository area. The current planned repository horizon is about as high in the section as it can go. One argument about the presence of drifting below the repository horizon was that it could constrain the ability to move the repository horizon downward. Accordingly, the design group recommended developing a drift above the repository horizon that could also be used as a performance confirmation drift. The design and testing groups reached the consensus location shown on Figure III-2.

The Panel is impressed by the thoroughness with which the ECRB work was accomplished and applauds this type of activity.

B. Site Conditions With Waste Present

Effects of Thermal Pulse on Analyzability of Repository Behavior

Introduction

In assessing the viability of the proposed repository at Yucca Mountain, it has become clear that the effects produced by the thermal field are a key problem in developing a creditable basis for moving forward with the TSPA. The central issue is to understand and predict, with reasonable accuracy, the impact of the thermal field on both the near field and the far field. The far field consists of the total rock mass extending from the surface of the land downward about 300 m below the surface where the proposed repository is to be constructed. The near field is the rock mass that is in the vicinity of, and includes, the repository's engineered barrier system. This system will be constructed with a massive array of tunnels and drifts in which canisters, with their various waste forms, will be emplaced. Predicting the thermal disturbance created by the emplaced waste on both the near and far fields is a formidable challenge and leads to a basic question: "Under these circumstances, how thoroughly and accurately can the effects of a thermal pulse on the behavior of the repository be analyzed?"

In addressing this question, a comprehensive program of analysis has been underway for some time, and a large number of reports on the results are now available (see below). A number of models that can simulate the physics and chemistry of the governing processes have been developed. In particular, the response of the proposed repository under: (1) the current ambient conditions, and (2) the impact of the thermal perturbation, has been analyzed at length. This has been an effort without precedent, and is complicated by the fact that sufficient empirical evidence on the thermohydrologic, thermochemical, and thermomechanical behavior in systems of this kind is not available. Under these circumstances, it is understandable that there will be uncertainties in the results. Those uncertainties must be explicitly recognized by the TSPA team and evaluated to the degree possible.

Uncertainty in Percolation Rate and Flux

The percolation flux at the level of the proposed repository in the middle of the non-lithophysal portion of the Topopah Spring welded tuff (TSw) is one of the most critical parameters both in interpreting the current site conditions and in assessing its suitability as a potential repository. Presumably, this flux has led to the present distribution of water saturations in the matrix of the TSw, which range from 50% to 70% in the top half, up to 90% to 95% in the bottom half, of this layer. This is where most of the proposed repository will be located (Bodvarsson and Bandurraga, 1996).

In analyzing the problem of predicting the percolation flux in the UZ, Wu et al. (1997) state that there exists a large number of uncertainties, key among which are: (1) sizable ranges for the estimated current, past and future net infiltration rates over the mountain; (2) large variances in the measured and calibrated tuff property sets; (3) spatially varying property distributions within the mountain representing lateral heterogeneities, especially for fracture/matrix parameters in the TSw unit; and (4) lack of confirmation of the mechanisms and a numerical scheme for fracture/matrix interactions in the welded units. Given that *in situ* percolation values in the mountain are difficult to measure directly, these investigators concluded that it will be difficult to calibrate or verify accurate values for these parameters.

A large amount of effort, primarily by workers at the USGS, has been devoted to the problem of infiltration from rainfall. The current conceptual model for infiltration is based on numerous measurements of water content profiles in shallow boreholes. Flint et al. (in preparation) have also developed a numerical model to help in these investigations. Rainfall, which currently averages 150 mm/year, is spatially heterogeneous due to variations in soil cover and topography, and it is also variable with time due to storm events (Hevesi et al., 1994). A significant thickness of alluvium can store infiltration and attenuate an infiltration pulse. Thus, infiltration is high on sideslopes and ridgetops, where outcrops are exposed and flow into the fractured volcanics can take place (Flint and Flint, 1994). Modeling studies in the 1996 UZ Model report (Bodvarsson and Bandurraga, 1996) revealed significant differences in the effects of the thermal field on the hydraulic behavior of the repository system as the input value for the infiltration rate was varied from the previous estimate of 0.1 mm/year to the current estimate of 4.4 mm/yr. The magnitude of this critical factor must be well established, if its effects on repository behavior are to be accurately evaluated.

In predicting the percolation fluxes at the repository, an adequate account of the hydrologic properties of the mountain, in particular the fracture/matrix interaction, is necessary (see also discussion below). To match the recently revised estimates for the infiltration flux (currently at 4 mm/year), Wu et al. (1997) were forced to introduce the concept of a fracture/matrix reduction factor that significantly reduces fracture/matrix interactions in the welded tuff layers. This effectively leads to a smaller lateral diversion of water in the model, and allows for physically acceptable estimates of hydrologic parameters consistent with field measurements.

Based on field data, the higher infiltration zones are located along Yucca Crest from north to south. High percolation fluxes at the repository horizon, however, are predicted to be located several hundreds of meters east of the high net surface infiltration area. If higher interactions between the matrix and fractures are assumed, the lateral diversion is significantly increased with important consequences on the distribution of the percolation flux at the repository horizon. Wu et al. (1997) established an upper limit for the average infiltration rates at Yucca Mountain as being no more than 15 mm/year, based on these studies.

Uncertainty from Treatment of Fracture/Matrix Interactions

As noted above, a major obstacle in model development has been the problem of characterizing and modeling the fracture/matrix interactions. Otherwise, the factors controlling the flow of fluids in these two components, with very different hydraulic parameters, cannot be handled correctly. In many cases, this interaction takes the form of a competition between advection in the fracture network and diffusion (mass, heat, capillarity) in the matrix. In particular, the partition of flow between fracture and matrix is dictated by parameters such as the capillary diffusivity (imbibition), the area of interaction and the maximum amount of trapped saturation of the non-wetting phase (air) in the grid block volume.

In the current coarse grid simulation (for example, using the dual permeability model (DKM), where both the fracture and matrix are modeled as distinct parts of the system), the representation of these interactions is through effective parameters, such as the area between fracture and matrix. As noted above, this is currently expressed through a reduction factor to reflect the limited contact resulting from channelized fracture flow. Reduction factors as low as 10^{-3} have been postulated to match field data. This is a drastic departure from the simulation practice only a year ago, where this concept was not used. Although the concept of a limited contact area correctly reflects the physics at the fracture/matrix interface, this factor is currently being used as a fitting parameter in an *ad hoc* fashion. Additional uncertainty, particularly for two-phase flow processes (imbibition, drainage and heat pipes), is introduced due to the volume averaging over a number of fracture-matrix areas, included in coarse grid blocks. In such cases, the set of hydrologic parameters used will not correspond either to that of individual fractures or matrix blocks.

With or without a reduction factor, the use of the DKM has only been partially successful in capturing the fracture/matrix interaction in thermohydrologic applications. It has not been possible to conduct investigations over long enough periods of time to reveal the complete picture of the impact of the thermal perturbation on the repository under an assumed heating load. Currently, this is done using the equivalent continuum model (ECM) in which it is assumed that thermodynamic equilibrium exists between fracture and matrix. On this basis, an appropriate averaging of coefficients can be used to obtain an effective continuum.

Based on an analysis of the fracture/matrix interaction in Appendix A, one can show that reaching conditions of fracture/matrix equilibrium is controlled by the magnitudes of the diffusivity, fracture spacing and flow rate. For typical conditions in Yucca Mountain, fracture/matrix equilibrium is likely for thermal energy and for the imbibition of a high-permeability matrix, but not necessarily for mass diffusion and the imbibition of a low-permeability tuff. The latter is common to many rocks at Yucca Mountain, and in such cases, the assumption of equilibrium will fail. ECM cannot also account for a fracture/matrix reduction factor; thus, it is inherently unable to match the revised percolation flux (unless a non-zero value for the trapped air saturation is introduced, which is not currently done). Nevertheless, the ECM model has been used extensively in investigations of the thermohydrologic behavior of the repository over very long periods of time.

A more detailed analysis of the fracture/matrix interaction is given in the report "The Fracture Matrix Interaction: Reduction of Uncertainty." This report, prepared by Y.C. Yortsos, who is a consultant to the Panel, is given in Appendix A. As noted, he raises a number of questions about the manner in which this subject is being analyzed. The Panel shares these concerns and makes the following recommendations to improve the state of the art in this subject area:

1. <u>Revisit the concept of reduction factor.</u> Use the experiments reported in Glass et al. (1997) and earlier publications, which give a wealth of information on the displacement patterns at various conditions, to estimate reliably the effective area (and the corresponding reduction factor). Then, account for a possible increase of this factor due to the stabilization of the displacement exerted by imbibition in the matrix. Modify the fracture hydrological parameters, particularly the relative permeabilities, to account for the fingered displacement, where appropriate, by considering rate and

gravity effects. Allow for anisotropy in permeability, displacement and reduction factor in the fracture continuum in the horizontal and vertical directions. In this context, reassess the effect of mineral precipitation at areas of geochemical interaction that are expected to occur in the near field (see related comments below).

2. Allow for the possibility of non-zero trapped (residual) air saturation. Account for non-zero trapped saturation in the various lithological units, by considering the direction and rate of invasion (imbibition). Consider the effect of large-scale trapping, due to large-scale heterogeneity in the grid block, in increasing the effective residual gas saturation. Non-zero values may lead to lower, and thus more defensible, reduction factors,

3. Improve the estimation procedure for matching field hydrologic data. Analyze the limitations of the one-dimensional model (only vertical flow) currently used to match field data and estimate parameters. Allow for the possibility of lateral flow, due to capillary and flow barriers, anisotropy, etc. Study the consequences of non-uniqueness inherent to the inversion process.

4. Improve the large-scale description of two-phase flow processes. Revisit the formalism for representing unsaturated now in a grid block, by accounting for effective large-scale permeabilities, relative permeabilities, capillary pressures, large-scale trapped saturations and the fracture-matrix interaction. In this context, particular attention needs to be given to the heat pipe description. Consider the extension of the particle-tracking algorithm to three-dimensional and other diffusive processes.

5. Justify the use of ECM for Thermal predictions. Carefully delineate the validity of capillary equilibrium in ECM applications. Revisit the ECM formalism and validity in light of 1 and 2 above, and also revisit the heat pipe representation (see below).

Uncertainties in Coupled Processes Driven by Thermal Disturbance

The thermal disturbance is expected to affect the hydrology, chemistry and mechanical response of the mountain, particularly in the near field. Thermohydrological coupling occurs mostly in the form of heat pipes; thermochemical coupling is manifested in the chemical alteration of the near field; and thermomechanical coupling produces rock displacements with the notable possibility of altering hydraulic properties, such as fracture permeabilities. Considerable uncertainties currently exist in the understanding and modeling of all these processes. In recognizing the need for the reduction of these uncertainties, a series of *in situ* thermal tests has been proposed.

The first underground thermal test conducted in the ESF is the single-heater test. The preliminary findings have some interesting implications with regard to the anticipated thermal response of the rock system in which the proposed repository may be constructed. A description of the test design, plans and layout area has been prepared (CRWMS M&O, 1996). The heating period for the Single Heater Test started August 26, 1996 using an electrical heater with an active length of 5 m and power input of ~3800 w. Rock temperatures in the near vicinity of the heater exceeded 100° C after about 20 days and were at about 160° C at the end of the 9-month heating period. During this experiment water collected in one instrument hole, and about 17 liters were saved for analysis.

Thermohydrological results

A preliminary analysis of the Single Heater Test results from the thermohydrological standpoint has been reported by Tsang (1997). Before proceeding with her findings, it should be recalled that as temperatures in the repository reach the boiling point, a heat pipe mechanism will set in (shown schematically in Figure III-3). For a fracture/matrix system, the conceptual model is that water vapor (steam) will reside mostly in the fracture, while condensed water reflux will occur mostly in the matrix due to imbibition, although the possibility of liquid counterflow in the form of films along fracture walls can not be discounted. Boiling and condensation processes above the heat source are not necessarily the same as those below (Figure III-3). Above the heat source, the extent of the heat pipe is larger as gravity aids in the return flow. Below the heat source, the return of condensate is only by capillary action, because gravity acts to move the liquid away from the source. In either case, the possibility exists that under the influence of gravity, flow in the fractures can lead to a loss of mass away from the source. This is another indication of the critical importance of properly understanding the nature of the fracture/matrix interaction. It is also evidence that the loss of mass can lead to difficulties in developing an appropriate numerical model of the system behavior.

In analyzing the Single Heater Test, Tsang (1997) states that good agreement was found between field data and simulations and suggests that the thermohydrologic processes of the heating phase are well understood. As others have reported for similar experiments, heat conduction is the main mode of heat transfer below the boiling point. However, an appropriate account for the effects of convection (and the fracture/matrix interaction) is necessary to predict the flow rates and locations of fluid mobilized by boiling (as evidenced by the water collected in one instrument hole). There was disagreement between model predictions and the measured temperature field (almost 30º C at places). Tsang attributes this to spatial heterogeneities which apparently were not detected in the pretest characterization work. She also indicated a problem of uncertainty in the hydrological properties being used, particularly the matrix and fracture characteristic curves. The Panel expects that the effect of this uncertainty will be amplified at later times in the test, now that cooling is taking place and re-wetting will occur. In analyzing the moisture redistribution, Tsang used both ECM and DKM models and reports that a better agreement was obtained using the DKM model for the asymmetry of the condensation zone surrounding the heater horizon. However, she did not make use of the fracture/matrix reduction factor, mentioned as an essential component of the work of Wu et al. (1997).

Being the first thermal experiment at the level of the proposed repository, it was of considerable interest to determine whether it might be possible to see some evidence of the effects of the ambient percolation rate. The thermally induced fluxes are orders of magnitude larger than the ambient flux, thus precluding the detection of the effect of ambient percolation (Tsang, 1997).

Thermochemical results

A preliminary analysis of the Single Heater Test results from the thermochemical standpoint has been reported by Glassley et al. (1997). They have analyzed the water samples above and found pH values ranging from 6.2 to 6.9. These values contrast with pH values of 7.1 to 8.1 for waters collected from matrix, saturated zone, and fracture samples. They attribute these lower pH values to a condensate-fracture-matrix interaction that results from the CO_2 concentration, which is elevated relative to normal atmospheric concentrations.

Glassley et al. (1997) are primarily interested in investigating the hydrothermal processes that drive mineral alteration. Key parameters for defining mineral alteration are: (1) dissolution and precipitation kinetics, (2) thermodynamics of homogeneous and heterogeneous equilibria, (3) flow pathways, and (4) flow rates. As temperatures in the rock walls of the repository drifts exceed the boiling point, the matrix water migrates to nearby fractures where vaporization and heat pipes develop. Figure III-3 illustrates the nature of the fluid movement. Water vaporization will lead to mineral precipitation at the fracture/matrix interface. Away from the heat source, in cooler regions, condensation occurs, and the condensate formation leads to chemical conditions that are not in equilibrium with the surrounding rock. Either of these geochemical interactions contains the possibility of altering the effective fracture aperture. The extent and location of these effects are dependent on the design and operation of the repository.

Glassley et al. (1997) have concentrated on developing an understanding of the nature and magnitudes of these processes. From modeling studies, they find that volume changes are possible as a result of dissolution in the condensation zone, formation of secondary minerals, and the involvement of the fracture and matrix in the chemical evolution. Carbonate, feldspar, and SiO_2 polymorphs can dissolve in the condensation zone, and clays and zeolites can precipitate along the flow paths. However, the extent to which these reactions can lead to significant changes in the porosity and permeability of the rock system is a major uncertainty at this point. Lin et al. (1997) have conducted laboratory investigations that indicate that the permeability of the fractured tuff could be reduced significantly. This may have significant implications on repository performance.

Thermomechanical results

Results from the Single Heater Test from the thermomechanical standpoint have been reported by Costin (1997). Although he has not yet been able to comprehensively analyze a very large data base, his preliminary evaluation of spatial and temporal variations of rock temperatures and rock deformations reveals the complications of analyzing the thermomechanical behavior of fractured rock in the TSw. As others have found (Witherspoon and Cook, 1979), one cannot assume that the system behaves like intact rock. Because of its heterogeneous nature, fractured rock creates a complex medium that cannot be analyzed using the theory of linear thermoelasticity. Much more work on this problem is needed in order to reduce the level of uncertainty in one's ability to predict: (1) thermomechanical behavior of the system, and (2) whether or not the permeability of the TSw rock mass will be adversely affected by changes in fracture apertures. Results from the Single Heater Test will be of great importance in carrying out the investigations planned as part of the Drift Scale Test that was started in December 1997.

As discussed below in connection with the drift scale test Blair et al. (1997) have developed a new method to estimate changes in permeability due to thermomechanical effects. Their results indicate that these effects may cause a significant enhancement in permeability.

Implications for Analyzability of Repository Behavior

This discussion has not touched on problems concerned with the engineered barrier. Nevertheless, it is the Panel's view that uncertainties in the thermal behavior of the repository, revealed by the difficulties discussed above, could lead to questions on alternative designs for the repository. For example, if the thermal pulse were eliminated, as would be the case if the waste were cooled for an appropriate period of time, the effects of waste heat could be reduced or eliminated. Presumably, the uncertainties in the projections of repository behavior would also be reduced. It is well known that the concept of cooling spent fuel through surface storage has been adopted in Europe. The Panel suggests that it would be prudent for DOE to be prepared for questions concerning the analyzability of the thermal behavior of the repository as presently designed.

Drift Scale Test

Introduction

The Drift Scale Test is an important experiment that will provide the first large scale underground investigation on the critical problem of the behavior of the TSw under the impact of the thermal field. The Draft Scale Test will simulate the thermal conditions that will be created by heat released from the waste and investigate the range and magnitude of the different effects in the fractured rock mass.

The Drift Scale Test was first described as an "Emplacement Drift Thermal Test" and is one part of an *in situ* thermal testing program (DOE, 1995) for Yucca Mountain. The Drift Scale Test is located at Station CS 28+27 just off the ESF main drift, at the elevation of the proposed repository in the middle of a non-lithophysal zone in the TSw. The thermal load will be created using electrical heaters placed in a 5 meter drift, 47.5 meters in length and supplemented by wing heaters on both sides along the total length of the drift (CRWMS M&O, 1996).

Objectives

The objectives (DOE, 1995) of this test are to:

- Examine the near-field thermal-hydrologic environment that may impact the waste package (i.e., liquid saturation in rock and backfill, room humidity, propagation of "dry" conditions, liquid drainage in fractures, chemical evolution of liquid flux, and changes in permeability);

- Provide a conceptual model and hypothesis test-bed for which thermal and coupled T-M-H-C models can be used to examine issues of heat transfer, fluid flow, and gas flow that will place realistic bounds on the expected nature of the near-field environment;

- Evaluate the effect of ground support interactions with the heated rock mass, including the effect of materials used for ground support on the near-field water chemistry;

- Measure corrosion rates on typical waste package materials under *in situ* conditions;

- Provide detailed measurements of the response of the rock mass to the construction and heating of an emplacement-drift-scale opening; and

- Provide bounding measurements on the thermal-hydrologic behavior of backfill materials.

Pretest Analyses

Birkholzer and Tsang (1997) have performed an interesting pretest analysis of the thermohydrological conditions for the Drift Scale Test. As part of this exercise, they assumed that the optimum heating schedule will apply almost full heater power in the first year to bring about a fast response in the Drift Scale Test, followed by a three-year period of reduced power output during which the rock temperatures are to be maintained at levels that do not exceed 200º C. It was assumed that the four-year heating period will be followed by a four-year cooling period.

Under these constraints, Birkholzer and Tsang (1997) have used two dimensional models to analyze the temporal evolution and spatial variation of the thermohydrological conditions in the rock mass and to evaluate the impact of different input parameters such as heating rates and schedules, and different percolation fluxes at the test horizon. They have also investigated the problem of the fracture/matrix interaction using ECM and DKM models, but as indicated above, the Panel is not convinced that the fracture/matrix problem is being properly handled in this work.

Another pretest analysis of the Drift Scale Test has been completed by Blair et al. (1997). This relates to the thermomechanical effects in the rock mass. The basic problem is the extent to which the rock permeability will change. Increasing stress across fractures causes a reduction in fracture aperture and a consequent decrease in flow through the fractures (Raven and Gale, 1985). The aperture is generally reduced as compressive stress across the fracture is increased. Thus, as the stress level in the potential repository horizon increases due to thermomechanical effects, the apertures of some fractures may be reduced with a consequent reduction in permeability of the rock mass. However, changes in the stress field may also increase shear stresses on favorably oriented fractures, leading to shear displacements and an increase in permeability (Olsson and Brown, 1994).

Blair et al. (1997) have developed a new method to estimate changes in permeability due to thermomechanical effects, and they present the results of a preliminary analysis of these effects in connection with the Drift Scale Test. Their results show that thermomechanical effects may cause a factor of 2-4 enhancement of the permeability over major regions of the heated rock. This enhancement occurs in the first few months of heating and may accompany the thermal pulse as it travels outward from the heat source.

A critical issue in the methodology linking the thermomechanical analysis to permeability is that permeability enhancement occurs as the result of shear offset due to Mohr-Coulomb slip on pre-existing fracture sets. In this study, Blair et al., (1997) used only two fracture sets in estimating changes in permeability, but the method can easily be adapted to three dimensions. This concept can be tested by comparing displacement measurements made during the Drift Scale Test with those predicted by their model. Unfortunately, the geometry of the wing heaters used in the Drift Scale Test introduces thermomechanical effects that may be much different from those that will be developed in the proposed repository where the heat sources will be located only in parallel drifts.

Conclusion

The Panel believes that the Drift Scale Test will constitute a major step forward in the process of understanding the complex behavior of the proposed repository under the impact of the thermal field. Despite the surprises that are bound to occur, a wealth of data and information will be gathered. An analysis of the results will provide a basis for determining the applicability of our present understanding of the controlling features of the thermal perturbation, as well as much needed data for model calibration. The Panel recommends that an open schedule be adopted for the length of time that the Drift Scale Test will be kept in operation. Underground testing in fractured tuff on this scale has never been done before, and a reduction of uncertainties is anticipated that will be important as DOE approaches the license application phase.

C. Engineered Barriers and Waste Package Performance

Introduction

An effective Engineered Barrier System (EBS) and a robust Waste Package (WP) are essential to the overall performance of the repository. The goal is to design the EBS and the WP for:

- A long isolation period to permit essentially complete decay of many of the radionuclides in the waste, and

- Controlled slow release of the remaining radionuclides to the adjacent geologic formation.

There continues to be significant progress by the TSPA team on the analysis of the EBS/WP performance; however, there remain major areas of concern that can have negative impact upon the TSPA-VA.

In this section, the Panel presents comments first on waste package issues and then on engineered barrier system and waste form/radionuclide release issues. This is not a comprehensive treatment, but rather it is intended to provide input to the TSPA team while work is in progress.

Waste Package Issues

Effects of water seepage

Depending on the extent of the thermal pulse and the response of the geologic system, seepage is likely to result in water/moisture coming into contact with some of the waste packages at some time. Since the amount and distribution of such contacts will have spatial and temporal variations, it is prudent to design the waste packages with the expectation that they will be contacted by repository waters. To the extent that the packages remain dry, the benefit can be considered defense-in-depth. Although steel barriers will sustain progressive and cumulative damage from each period of wetness, corrosion resistant metal barriers that remain passive will exhibit essentially no attack (0.1 to 1 micron/year) during wet periods.

Crevice corrosion of the corrosion resistant metal (CRM) barrier is the primary degradation mode to be avoided. Alloy C-22 and titanium are resistant to localized corrosion in the nominal repository environment as well as in many environments beyond this range. The determination of a realistic range of environments that can contact the CRM barrier is the critical requirement for understanding the performance of the waste packages.

The determination of water seepage into the drifts is a matter of large uncertainty. The treatment of water seepage onto waste packages in the TSPA-VA is based on the determination of distribution functions for seepage over the population of packages and additional distribution functions of seepage over individual packages. Those members of the TSPA team responsible for developing the seepage functions must deal with spatial and temporal variability. The combined functions are used to turn-on and turn-off the wet corrosion of packages. The more resistant the packages are to damage from water seepage, the less impact the uncertainty of water seepage will have on the analysis of overall repository performance and reliability**.**

Metal selection for inner barrier

The reliability of the TSPA-VA is increased and uncertainty is reduced by the selection of highly corrosion resistant metals for the waste packages**.** As the Project team has progressed in the design of the proposed repository, the use of more corrosion resistant materials, i.e. Alloy 825 to 625 to C-22 and titanium, has been proposed. Alloy C-22 (a high nickel-chromium-molybdenum alloy) and titanium represent two of the most corrosion resistant classes of metals in oxidizing-chloride solutions (the most prevalent wet environment anticipated in the repository). Such a proposal is prudent for several reasons: (1) resistance to localized corrosion is required for long term containment; (2) there is considerable uncertainty in the prediction of the range and chemical composition of the localized waters in contact with the waste package; and (3) water contacting the waste package should be assumed for this portion of the design. For these reasons, the Panel supports these actions.

In the opinion of the Panel, the designation of the alloy for the corrosion resistant inner barrier of the waste package should be considered a “place holder” that represents an alloy of a given class of metals, e.g. highly corrosion resistant, nickel-chromium-molybdenum alloy. Other specific alloy designations with equivalent or better properties can be expected to provide comparable service.

Effects of crevice corrosion

The Panel concurs with the conclusion of the Waste Package Expert Elicitation (“Waste Package Degradation Expert Elicitation Project Final Report, ” August 15, 1997) that crevice corrosion is the most important degradation mode to be considered in the TSPA-VA**.** Such corrosion of the corrosion resistant metal results from the localized breakdown of the protective (passive) film on the metal. Crevice corrosion is more aggressive than pitting, and a material selection based on crevice corrosion resistance is both more realistic and more conservative. Crevices will always occur and cannot be completely avoided anytime there is contact involving metal/metal, metal/EBS material, metal/rock, metal/corrosion product or deposits. The corrosion control approach is to: (1) determine the range of corrosive environments that pertain; and (2) select materials that are resistant to crevice corrosion in those environments.

The nominal environment in the repository, i.e. neutral to mildly alkaline carbonate waters with low levels of chloride, is not aggressive to corrosion resistant metals at temperatures up to the boiling point. The concern is with modifications to the nominal conditions that arise from the thermal pulse in the rock, interaction with EBS materials, corrosion products, and later on with materials within the packages.

Microbial activity in the drifts is another process that can affect the water composition in contact with the metals; however, it is unlikely that microbial activity will extend the corrosive conditions beyond the range already being considered. Furthermore, the highly corrosion resistant metals being considered are not affected by microbially induced corrosion (MIC).

Environments in contact with the waste package

There is a paucity of experimental data to support either the selection of materials for the waste package or to test and validate the models for assessing their performance. Experimental approaches and methods to determine crevice corrosion environments are well established and do not require long test times. The Project and TSPA teams should exploit these opportunities.

Corrosion resistance of metals

Experimental approaches and methods to determine crevice corrosion resistance are well established and do not require long test times. The Panel recommends that tests be run to determine the behavior of C-22, Ti, 625, and 825 in a range of environments not only to cover the expected repository conditions, but also to extend well beyond these conditions. The inclusion of the less corrosion resistant metals and the more corrosive environments will provide a measure of the margin provided for unexpected conditions.

There is clear agreement among corrosion science and engineering specialists as to the effect of environmental conditions on the occurrence of crevice corrosion, and there is agreement on the relative effectiveness of the metal alloys in providing resistance to crevice corrosion. Unfortunately, there is a lack of experimental data from the project on the behavior of the alloys of interest under realistic repository environments. Notional information is available; realistic data are needed.

The current status can be summarized by a notional figure presented in material prepared for the Waste Package Expert Elicitation Panel and presented at the NWTRB Meeting Oct. 23, 1997. This figure (see Figure III-4) below) presents the relative resistance to crevice corrosion for steel, Alloy 625, Alloy C-276, and Alloy C-22. The last three are nickel-chromium-molybdenum alloys with increasing corrosion resistance in the order presented. For purposes of presentation, the notional crevice corrosion resistance is plotted versus the corrosive environment. On the lower horizontal axis, increasing oxidizing power of the environment is shown as more positive electrochemical potential. The upper horizontal axis shows the notional positions of an oxygen containing environment (O_2), an environment with active microbial activity, and a highly oxidizing environment containing ferric ions (Fe^{+3}). The S-curves show the boundary between no corrosion (to the left) and the initiation of crevice corrosion (to the right). The dashed lines are the notional representation of uncertainty for corrosion behavior. Data generated by experiments are required to support materials selection and assessment of realistic performance.

Useful data are available from the published literature. These data demonstrate the high level of crevice corrosion resistance of C-22 and titanium. For example, the critical crevice corrosion temperature for C-22 is given as 102° C in an oxidizing acid with high concentrations of chloride (pH 2, 4.3% NaCl) (Gdowski. 1991). This is a highly aggressive environment far from the nominal conditions of repository waters.

Dual CRM packages vs. Steel/CRM packages

The reference case for TSPA-VA is likely to specify a dual-canister waste package comprised of a steel outer layer (corrosion allowance metal) and a nickel-chromium-molybdenum alloy inner layer (corrosion resistant metal). The attributes of this design have been well defined by the Project team. A steel outer barrier has several desirable features that would be useful, particularly during a long, dry period. When wetted, however, a steel canister will corrode rapidly. Because of the complex interactions of iron corrosion products on the chemical and mechanical processes within the drifts, this will increase the uncertainty regarding the response of the inner barrier. Dual packages comprised of a double layer of corrosion resistant metals, e.g. C-22/titanium or titanium/C-22 have been proposed and are worthy of further consideration and evaluation in the performance assessment

The temperature limit within the waste package

In order to protect the zircaloy fuel cladding from rapid deterioration, the Design team has specified 320° C as the upper temperature limit with in the waste package. Above this temperature, zircaloy is subject to creep rupture. There are likely other sound reasons to maintain this as an upper temperature limit. These include the fact that there is a wide range of heat output from the spent nuclear fuel (SNF) and a variety of placement configurations within waste packages. Is 350° C the upper limit (e.g. 99th percentile) of the waste packages? Is 350° C the hottest area within a waste package, and what is the average temperature over the waste package?

It is also not clear to the Panel how this limit will be treated conceptually. The heat source has major impacts on many processes within the repository. High heat output increases the duration and extent of the dry out period. A beneficial result is that the longer duration of dry conditions will forestall the onset of wet corrosive conditions. Conversely, the elevated magnitude of the thermal pulse will increase the effects of the geological site on overall repository performance and increases the uncertainty regarding the thermal-hydrological response.

Corrosion data and service experience

The durability of the canisters of the waste package and their likely times-to-penetration have been shown to have a significant effect upon the TSPA results. A long-lived canister has an important and positive effect. All of the available information from the literature and service experience regarding the corrosion behavior of the corrosion resistant metals should be gathered to support materials selection for performance assessment.

The Panel recommends that a comprehensive compilation and critical review of the corrosion behavior of the two primary candidates for the corrosion resistant metal (CRM). These efforts should be directed to the two classes of alloys, namely, nickel-chromium-molybdenum alloys and titanium alloys, and not to a specific metal designation. Earlier efforts (e.g., Gdowski, 1991) should be updated and expanded. The scope of the review should include laboratory data and service experience, as well as information on metallurgical stability and the effect of welds (microstructure and composition). These compilations provide guidance and focus to project experimental needs to validate materials selection and performance, but they do not relax the need for project specific data**.**

Corrosion rates relevant to passive metals

The time-to-penetration of canisters of the waste package is an important factor in the TSPA analysis, and it has a major impact on the calculated repository performance. The corrosion rate that is used when the corrosion resistant metal canister is wet is a fundamental parameter in the TSPA. The values determined for the corrosion rates and the level of confidence in these values being realistic will have a critical effect on the evaluation of the TSPA-VA.

Penetration rates as low as 0.1 to 1 micron/year are not unrealistic for corrosion resistant metals in the passive state. Such penetration will be fairly uniform and projected penetration rates of 10,000 to 100,000 years/cm of CRM result. When crevice corrosion is active, the metal penetration rates are high and rapid penetration can be observed (1 to 10 mm/year). Clearly, confidence in the long term performance of the corrosion resistant barrier depends on the selection of metals which, under the anticipated environmental conditions, will provide high resistance to crevice corrosion.

In short, the need is to select materials that will realistically remain passive in the repository for long periods of time. First, it is necessary to document that the corrosion resistant metals have a high resistance to the initiation of crevice corrosion. Furthermore, it is necessary to document that should crevice corrosion initiate there is a high propensity for arrest of the corrosion and a return to the passive state. A structured experimental program and modeling effort to address both issues above are required. In addition to determining the metal/environment behavior regarding crevice corrosion, It will be necessary to develop a rationale for the behavior with respect to chemical and electrochemical processes.

Although the Project team appears to be moving in this direction, the current plans do not fully address these issues. The work in Canada on titanium corrosion for waste storage (as presented to the Waste Package Expert Elicitation Panel) provides a useful guide and approach.

Stress corrosion cracking

No mechanistic models for stress corrosion cracking (SCC) are available for TSPA-VA, and it is not recommended that project resources be allocated for stress corrosion modeling. Rather, an engineering approach is recommended to select metals that are resistant to SCC and to specify design and manufacturing procedures that avoid SCC.

Stress corrosion cracking is a threat to the adequacy of waste package performance. Full penetrations result in short times if SCC occurs . For a given metal, the environmental conditions and magnitude of tensile stresses control SCC. The required approach is to select materials that are resistant to SCC in the anticipated repository environments and to avoid tensile stresses to the extent possible. It is not practical to design for arresting stress corrosion cracks once they have begun, because the crack growth rates are too rapid compared with the long life desired for the waste package. This leads initially to the selection of materials that are highly resistant to crevice corrosion. Once these materials have been identified, consideration needs to be directed to how they will resist conditions that could lead to SCC. The previously cited concerns regarding the uncertainties and lack of experimental data for environments anticipated to be in contact with the waste package also pertain here.

Control of tensile stresses to avoid stress corrosion cracking is a fundamental part of the required design strategy. Tensile stresses cannot be completely avoided; however, the manufacturing, handling, and service conditions can be reviewed and evaluated to select material and maintain conditions so as to minimize stresses. Residual stresses from cold work, differential thermal expansion and welding are the most important. Rock falls can also be a source of residual stresses to the packages after emplacement. From the perspective of undesirable tensile stresses, the proposed shrink-fit operation and welds without subsequent stress relief are of most concern.

Effects of corrosion products

Gaps will exist between the CRM inner barrier and the proposed steel outer waste canister barrier. Once the integrity of the outer barrier has been lost, water can penetrate these gaps along and around the waste packages and this can lead to the growth of corrosion products in these gaps. The corrosion products of steel will occupy more space than the parent metal. As corrosion progresses, the gap will be filled. Further expansion will apply loads to the canisters that can be sufficiently high to deform the metal. Two practical cases of this phenomena are “pack out” damage to bolted steel structures and “denting” in PWR steam generators.

Shrink fit of inner and outer waste package canisters

The shrink fit process involves heating the outer barrier so that it expands, and then lowering it over the inner barrier where it contracts on cooling to give a tight fit between the two canisters. While this is desirable from some perspectives, the potential effects and implications of this process introduce additional uncertainties. First, the residual stresses resulting from the process need to be considered with respect to stress corrosion cracking. Secondly, a thick iron oxide coating will form on the steel surface after it has been heated. This oxide layer will remain in the crevice between the two barriers after the shrink fit assembly is completed. The potential effects of the oxide coating on waste package performance must be considered.

Galvanic protection

As discussed in the report on the Waste Package Expert Elicitation, the extent to which galvanic protection to the corrosion resistant barrier is provided by the steel outer barrier will be limited to the order of millimeters. The beneficial effect is realized from the shift of the electrical potential of the CRM to more active potentials below that which is critical for crevice corrosion. When the corrosion potential of the CRM is more negative than the critical potential for crevice corrosion, no galvanic protection occurs. The need for and extent of the required galvanic protection will depend on the geometry of the galvanic couple, the degree to which the outer barrier has been penetrated, the resulting exposed area of inner barrier, the presence or absence of corrosion products and deposits and the chemical composition of the waters present. The basis for any credit/benefits for this type of protection in the TSPA-VA must be explicitly presented and documented.

Engineered Barrier System and Waste Form/Radionuclide Release Issues

Conceptual drawings of EBS/WP over time

The development of schematic drawings and notional figures of the appearance of the EBS and waste package at various times are extremely helpful in understanding the various design configurations being considered. These are especially useful in conveying the expected results. The Panel encourages further development and refinement of these approaches.

The long dryout period

An extended dryout period resulting from the heat output from the waste packages is a basic feature of the current design. The extent of the dryout period is determined by the heat output from individual packages, the placement of packages along the drift and the spacing between drifts. As previously noted, the thermal pulse will not be uniform due to variations in packages, package placement, unused or unusable areas within the repository, and edge effects around the repository. This will affect the movement of water to and away from the drifts. The Panel recommends that increased effort be made to develop the conceptual description of the response to this large and nonuniform thermal pulse.

Chemistry of waters entering the drifts

The nominal water chemistry in the unperturbed repository is a mildly alkaline (pH 9), dilute (10^{-3} molar) bicarbonate solution with low concentrations of chloride, sulfate and silicates. The gas in the repository is essentially air with modest increases in carbon dioxide. The rock and waters will be heated by the waste packages, and the thermal pulse can extend into the rock for distances up to tens of meters from the drifts, depending upon the density of thermal loading. As the water is heated above boiling, a water vapor plume will extend from the area of the waste emplacement out into the rock. Many of the thermal, hydrological and geochemical processes have been identified. However, as mentioned above, the conceptual description of the thermal pulse effects is poorly developed and more effort needs to be directed to an evaluation of its impacts.

Large volumes of water are mobilized by the thermal pulse. The flow paths and amounts of water transported along various paths are not well defined. This leads to large uncertainties regarding the amounts and distribution of seepage flowing back into the drifts. The spatial and temporal flows are uncertain. From the perspective of reducing the uncertainties in rate of waste package corrosion, the Panel notes that essentially no damage will occur during dry periods for steel or the corrosion resistant metal barriers. Steel corrodes rapidly when wet, and cumulative damage will occur during intermittent wet periods. The corrosion resistant metal should be selected to remain passive when wet, so that extremely low corrosion rates can be obtained.

The water chemistry of heated water has been modeled by Glassley and others, and there are limited experimental data to serve as input to these models. Current models do not correlate well with experimental observations. More experimental data (laboratory and field) are needed to determine the water chemistry under realistic conditions and to refine and validate the water chemistry models. Early results from one of the heater tests indicate that water flowing due to heating were more dilute and less alkaline (pH 6-7) than cooler waters. Carbon dioxide in the gas phase was increased from the unperturbed conditions.

None of these waters (perturbed or unperturbed) is corrosive to the CRM inner barrier. Conditions conducive to corrosion require the presence of either highly acidic, high chloride solutions or highly alkaline solutions. No realistic conditions to generate these corrosive waters have been demonstrated for the proposed repository; however, the realistic range of water compositions in contact with waste package metals is yet to be determined.

Modification of water chemistry by concrete

The pH of solutions in contact with concrete will become alkaline due to reaction of water with concrete structures in the drifts and this process may affect water chemistry in the drifts. As the concrete degrades by carbonation, it loses its ability to release alkaline species. The condition and distribution of concrete during the period when water enters the drift and the water pathways are uncertain. The amount of water that enters the drifts will affect the extent of this affect and the duration over which it operates. Some clarification of this issue is needed.

Modification of water chemistry

The potential for modification of water chemistry, while in and on egress from the waste packages, remains an area of major uncertainty. The current project strategy and activities are unlikely to determine a realistic set of water chemistries for water entering the drifts. The determination of water chemistries once a package has been penetrated is more uncertain**.** Once waters have entered the waste package through penetrations in the corrosion resistant metal barrier, they will encounter a wide range of spent fuel, cladding and internal assembly materials. It is unlikely that any current model will reliably predict realistic water chemistries. Relevant experiments could be done to determine the water chemistries under a range of realistic conditions. Experimental work and models focused on the critical species are required.

Transport from the Engineered Barrier System

The conceptual description of transport from the EBS is poorly developed. The many processes that can occur have been identified by the Project team, but a realistic description has not been presented of the alternative transport modes and how they are distributed over a given waste package, over the population of packages, or over time. A critical factor is the form and amount of water transported into and from perforated packages. Water is the medium of advective and diffusive transport for radionuclides as soluble species and colloids.

There are major uncertainties regarding: (1) the number and distribution of penetrations through the packages; (2) the morphology of penetrations; (3) the presence or absence of corrosion products or deposits in the penetrations; (4) the form and composition of corrosion products/deposits outside of the penetration; and (5) the form and composition of waste form, transformation products and other materials within the package. In addition, the radionuclide forms, amounts and distributions are uncertain. These uncertainties have led to a treatment in the TSPA analysis that is unrealistic and likely to be overly conservative. For example, past TSPA analysts have assumed that all of the waste form is instantly wetted when the first penetration occurs.

Treatment of Spent Fuel Cladding

The long term performance of the cladding on spent fuel can have a significant effects on the exposure and release of radionuclides. Zircaloy has excellent corrosion resistance in a wide range of solutions, and its barrier performance is worthy of analysis. However, there are major uncertainties to be considered in the analysis. These include the condition of the cladding on arrival of the spent fuel at the repository site, the condition of the cladding when barrier performance is required (hundreds and thousands of years after emplacement); and the determination of the corrosive environment in contact with cladding after waste package penetrations. Neither Sweden nor Canada, two other counties that have announced plans to dispose of spent fuel, take credit for cladding in the analysis of their repository performance. The Panel recommends that the basis for any credit and treatment of this credit in the TSPA-VA be explicitly presented and documented.

Treatment of Backfill

It is the Panel’s understanding that the base case for the TSPA-VA will be the “no backfill” case. Nevertheless, the Panel recommends that, because backfills of various types are under active consideration by the project, an analysis of the backfill case be included to the extent possible in the TSPA-VA and that a thorough analysis be prepared for the subsequent TSPA for a possible license application. The objectives of performing such a backfill case analysis should be to:

- Determine if there are any phenomena that are qualitatively different from the “no-backfill” case and that may have been overlooked to date; this would be in contrast to learning that the major differences represent small quantitative differences in various parameters such as temperature, saturation, etc.

- Determine which experimental data, not now available, are necessary to perform the analysis of the “backfill case” properly in the longer time frame (over several years beyond the VA).

An initial "backfill case" analysis undertaken over the next few months might reveal the need for either modifying the drift-scale test that is just being initiated, or undertaking another test series that might take a substantially different direction.

To meet the two above objectives, the "backfill case" analysts need over the next several months to focus on identifying the key controlling features of the system with backfill, rather than launching a full-scale multi-year project that would ultimately complete the backfill-case analysis in more detail. In other words, the Panel believes that the proper approach is to "scope out" the issues at this early stage and to provide a sound technical basis to launch a full-scale analysis of the backfill case.

D. Waste Form Degradation and Radionuclide Release

Introduction

The Panel continues to review the models that will be used in the TSPA-VA to describe waste form corrosion and radionuclide migration. In the first interim report, the Panel offered preliminary comments on models to be used for spent fuel corrosion. In this second report, the behavior of the glass waste form is considered.

Although the Panel has continued to meet with principal investigators and DOE contractors (meeting at Argonne National Laboratory on November 14-15) to review waste form degradation models, we note that there is an on-going Expert Elicitation Panel which is addressing this topic; therefore, the following comments should be considered as preliminary until the final report of the Expert Elicitation Panel is available (March, 1998) and the final selection of corrosion/release models has been made for the TSPA.

Grambow (in press) has noted that the alteration mechanisms of high-level radioactive waste (HLRW) glass and spent nuclear fuel (SNF) are quite different. Glass is an aperiodic, thermodynamically metastable, covalent/ionic solid whose degradation depends on ion-exchange, surface complexation and Si-saturation. The UO_2 of spent nuclear fuel is a crystalline, redox-sensitive semiconductor whose dissolution behavior is mainly governed by redox mass balance at the oxide-solution interface. Thus, the corrosion of the spent fuel is very sensitive to radiolytic effects at the solid-liquid interface. For both phases, corrosion is accompanied by the formation of alteration phases (gels and crystalline solids) which may incorporate various radionuclides into their structures by precipitation, coprecipitation and sorption.

Glass Waste Form

Although the vitrified, defense waste will occupy a large volume (approximately 6,000 canisters), it will represent only 4,400 MTHM (equivalent) of the total 70,000 MTHM of the repository capacity. The vitrified waste will account for only five percent of the total activity, and most of this will be associated with short-lived fission products. Still, the total amount of radioactive material in the vitrified waste is substantial (approximately 10^9 curies).

As a result, the impacts of the corrosion of the vitrified waste could represent a significant source for potential releases of radionuclides from the repository This has been discussed in a system-level performance assessment (Strachan et al., 1990) which compares releases from spent nuclear fuel and vitrified waste. This study distinguished between radionuclides of low and high solubilities. For those of low solubility, the release from spent fuel packages exceeded the release from glass waste packages by a factor of two. For radionuclides with high solubilities, matrix dissolution controlled long-term release. In this case, the initial release of radionuclides from the gap and grain boundaries of the spent fuel dominated short-term release by several orders of magnitude, but the long-term release depended on the relative long-term dissolution rates for vitrified waste and the spent fuel (Strachan et al., 1990). Grambow (in press) has also compared the kinetics of the long term rates for these two waste forms and noted that the long term rates depend critically on two different phenomena: (1) for glass, the rate is related mainly to processes associated with silica “saturation” and (2) for spent nuclear fuel, the rate is most directly related to radiolytic, oxidative dissolution. For radionuclides for which concentrations are bounded by solubility limits, both the spent nuclear fuel and the glass will be contributing (at different rates) to the radionuclide inventory of the solution; thus, one must anticipate chemical interactions between these two very different waste forms, and the assemblage of alteration products which control solubilities may depend on this interaction.

The “Methods & Assumptions” Report of the TSPA-VA (CRWMS M&O, 1997a, pages 6-80 to 6-97) describes the approach taken in modeling the degradation of both the SNF and the vitrified HLRW. Expanded descriptions of the models for glass dissolution and radionuclide release are provided in the Waste Form Characteristics Report (Version 1.2, December, 1996) and a Lawrence Livermore National Laboratory (LLNL) Report (O’Connell et al., 1997). The basic approach is to develop a response surface that describes the dissolution rate for which the principal parameters are temperature, pH, and dissolved silica concentration. The input for the model will be experimental data provided by Finn and Bates (Argonne National Laboratory, but no reference given). The model will not consider other aspects of the solution chemistry.

On the basis of its review to date, the Panel makes the following preliminary observations:

1. The decision to use a response surface (based on a limited experimental data set) for the description of glass degradation and radionuclide release fails to take into account a large quantity of published laboratory data, the variety of conceptual models for glass dissolution, and the studies of natural analogues of glass dissolution which have been developed over the past twenty years. Although the response surface approach may be computationally efficient, glass dissolution can certainly be based on a mechanistic model which can provide a stronger basis for long-term extrapolation.

2. Because of the extensive amount of previous work on glass dissolution and the data available in the literature, one must reasonably expect that the TSPA-VA will include rigorous comparison of these data sets to the modeled response surface.

3. It is unclear to the Panel how models, which only have pH and silica concentration as their principal parameters, can be used to calculate solubility limits for phases that form during the alteration of the glass. The phases that form will be a result of groundwater/spent fuel/glass/canister material interactions. This will certainly depend on the evolution of the near field environment, an important issue identified at the Waste Form Degradation and Mobilization Workshop.

4. One of the important issues identified at the Waste Form Degradation and Mobilization Workshop was the time dependent evolution of solution compositions and the structure and composition of the alteration/gel layer on the surface of the corroded glass. This was also identified as an important issue in the workshop entitled, "Glass: Scientific Research for High Performance Containment" sponsored by the French CEA in Mejane-le-Clap in September 1997. The reason that the gel layer is now viewed as important is that it can either be an efficient "sink" for rare earth elements and actinides or a source of colloids with high actinide concentrations. The importance of the leached layer is illustrated in Figure III-5. More than 90% of the actinides may be concentrated in the leached layer. Although proper evaluation of the role of the leached layer and the effects of alteration products will require more information than is presently used in the TSPA, the potential retardation of actinides in this layer may justify a more sophisticated approach that considers the role of the gel layer.

5. Prior to the breach of containers and contact with water, the glass will experience an extended thermal pulse and be subjected to high fluxes of ionizing radiation that will reach saturation values during the first few hundred years of storage (Weber et al., 1997). The TSPA should determine whether there are any deleterious effects on the glass waste form as a result of the combined effects of heat and radiation prior to contact with water.

6. Reaction rates for glass dissolution increase with temperature. Has the TSPA evaluated the effect of reduced temperature (disposal away from the spent fuel assemblies) on the release rate? If not, the Panel recommends that they do so.

7. The present model does not explicitly include vapor phase alteration of the glass. Is this not the most likely form of alteration that will occur? Will the vapor phase alteration increase or decrease the durability of the glass when it comes into contact with aqueous solutions? In later sections of the "Method & Assumptions" document, reference is made to the abstraction of the "DHLW Glass Degradation and Radionuclide Release Model." This will include a consideration of the extent of vapor hydration prior to liquid water content, but it is not clear how this potentially important factor will be incorporated into the model.

8. The corrosion rates and reaction progress for glass are sensitive to glass composition (Ebert presentation, Argonne National Laboratory, November 14, 1997)(Strachan and Croak, in press). Will the use of a single glass composition in the TSPA-VA properly bound radionuclide release for the variety of glass compositions that will finally be disposed of at Yucca Mountain?

9. The model used to describe the dissolution of the glass waste forms does not account for concentrations of chemical species in the corroding solutions which may enhance the leach rates. A principal concern is the role of ferric iron released by corrosion of the steel canister of the waste package. Precipitation of iron silicates can prevent the solution compositions from reaching silica saturation values that result in a decrease in corrosion rate of the glass. The iron can also act as a sink for sorption of actinides on colloids which may either be mobile or immobile. The Panel calls attention to this issue which was raised in the 1995 Audit Review by the Nuclear Regulatory Commission (Baca and Brient, 1996). In the Panel's view, this issue still requires attention.

In a broader sense, such a comment emphasizes the need to consider the near-field environment as an integrated system in which spent fuel, cladding, glass, and canister materials interact with water that has reacted with near-field rock and concrete. This is a complicated geochemical system.

Closing Commentary

In a recent review of source terms used for spent nuclear fuel and HLRW glass in performance assessments, Grambow (in press) has posed a number of questions that should be addressed to waste form modeling in the TSPA-VA:

1. Is the relation between experimental data and model unambiguous? Are alternative models possible?

2. Is the mechanistic understanding of the corrosion process sufficient to allow for 'best estimate' extrapolation?

3. How can short-term (up to years) laboratory data be scaled to long-term processes?

4. Are the important, inherent uncertainties quantifiable?

This Panel echoes these questions.

E. Transport

Colloids

The transport of actinides in natural geologic systems can be either as dissolved species complexed with anions or as colloids. The concentrations of the dissolved species in solution can be estimated or at least bounded by a knowledge of the solubility limits of the expected, dominant solid phases. To the extent that solution concentrations are in equilibrium with the solid phases in the system, these concentrations are expected to remain constant over time, and the total release of radionuclides depends on the volume of water in contact with the waste. In the case of spent nuclear fuel, the solids which limit solubility concentrations are the original UO_2 in the used fuel and resulting uranium-bearing alteration products. These phases are expected to control uranium concentrations in solution. Other elements can be expected to have their concentrations limited either by the solubility limits of phases in which they are important constituents or by phases into which they are incorporated in trace amounts.

In general, the solubility-limited actinide concentrations are expected to have relatively low values; however, colloids provide a demonstrable way of maintaining elevated concentrations of actinides in solution, and colloids provide a demonstrable means of transport, e.g. as aquatic colloids which are ubiquitous in natural systems (Kim, 1991, 1994). In addition to the ability of actinides to form intrinsic colloids or to be sorbed onto mineral surfaces and form aquatic colloids, the dissolution and degradation of the waste form itself may serve as a source of colloids. Bates et al. (1992) have shown that the laboratory "weathering" of a prototype nuclear waste glass leads to the concentration of nearly one hundred percent of the Pu and Am into the colloid-sized particles in the alteration layer of the glass. Additionally, actinides sorbed on colloids may be transported at a faster flow rate than the solute species (Savage, 1994). Thus, the failure to consider colloid transport can lead to a significant underestimation of actinide transport (Ibaraki and Sudicky, 1995).

On the other hand, natural colloids may disassociate as solutions become more dilute or be filtered and trapped during transport through porous media. In his presentation to the Saturated Zone Expert Elicitation Panel, Professor D. Langmuir suggested that the fate of colloids could include:

- They are filtered out by crushed tuff backfill under unsaturated conditions.
- Intrinsic colloids, such as Pu-oxy-hydroxides, will degrade in undersaturated solutions as they move away from the waste and once in solution tend to be adsorbed by rock surfaces in fractures especially in the matrix.
- Actinides on the surface of geocolloids will tend to desorb with groundwater flow and to be re-absorbed by surrounding rock surfaces which have unoccupied sites and orders of magnitude more reactive surface sites.

On the other hand, the Nuclear Regulatory Commission has identified a number of critical technical issues relevant to colloid transport (Manaktala et al., 1995). Principal among these are:

- The identification of geochemical conditions that would inhibit particulate and colloid formation.
- The effects of the degree of saturation on geochemical processes, such as colloid formation and sorption, on the transport of radionuclides.
- The parametric representation of retardation processes.

Thus, there appears to be a rather wide range of views as to the importance of colloid transport on repository performance. Although it is not possible (nor necessary nor appropriate) for the Panel to summarize previous work on colloids, it is perhaps worthwhile to note the challenges inherent in modeling colloid transport. Kim (1994) has commented on the extent to which predictive modeling is now successful in describing colloid transport:

> Various approaches have been tried for formulating predictive modeling for the colloid-facilitated actinide migration and the aquatic colloid migration. Since too many assumptions are incorporated into these models, their applicability to real natural systems is still far from straightforward.

Further, in a summary of the role of colloids in transport, Savage (1995) notes,

> To date, this [colloid transport and dispersal] is poorly understood (although both laboratory and field data regarding colloid and groundwater chemistry are available), and there have been few attempts to incorporate such information into a dynamic colloid migration model able to quantify the impact of colloids on radionuclide breakthrough.

Finally, the fundamental analysis of the role of colloids in actinide transport depends critically on the knowledge of, and assumptions concerning, sorption of actinides onto free and immobile colloids. At present, this behavior is generally captured by the use of bulk K_d data; however, the limitations of such an approach are becoming increasingly evident as more experimental work is completed (Geckeis et al., in press).

Although the TSPA-95 report (CRWMS M&O, 1995) did not include a consideration of possible mobilization and transport of radionuclides by colloids, the report does include a discussion of colloid transport and a brief review of models that could be incorporated into the TSPA. The conceptual representation of models treats sorption of radionuclides onto colloids by the use of a distribution coefficient, K_d. Despite the apparent computational simplicity of the approach, one may anticipate a number of problems:

- Definition of the types and amounts of colloid particles.
- Definition of the number of sorption sites.
- Distinction between reversible and irreversible sorption.
- Definition of mobile vs. immobile colloids.
- Use of experimental data to estimate the above parameters.
- Scale-up of experimental data to field-scale models.
- Confirmation of field-scale models.

The "Methods and Assumptions" report (CRWMS M&O 1997a) discusses colloid formation and transport in two sections: (1) as part of the near-field geochemical environment (6.3); and (2) as part of transport in the unsaturated zone (6.7). In both sections, the focus of the discussion is a description of models that will be used to evaluate the significance and effects of colloid transport; however, little mention is made of the theoretical and experimental basis for these models. It would be useful to address some of the fundamental questions:

1. Will colloids form?
2. What types of colloids will form?
3. Will the colloids be stable during transport?

Without convincing answers to these simple questions, the models will be of limited use. Given the previously cited comments, the Panel is concerned that the TSPA team not be overly optimistic in what can be modeled in a convincing and defensible manner. The Panel notes that there appears to be an extensive data base from work at the Los Alamos National Laboratory (LANL) (Triay et al., numerous cited reports); however, there is only a very limited discussion of how this work (conceptual models and data base) will be used in the TSPA-VA. The TSPA team should anticipate that this subject will be given careful attention and scrutiny.

Recently, colloid (<1 micrometer size particles) transport has assumed increasing importance with the report of evidence for colloid transport of radionuclides through fractured volcanic rock at the Nevada Test Site (Kersting and Thompson, 1997). The Panel received an oral presentation from A. Kersting on this subject on November 10th. The data presented supported the contention that radionuclides (^{60}Co, ^{137}Cs, Eu, Pu) are concentrated in the colloid-sized fraction; more than 90% of the measured radioactive material was detected in the particulate and colloid sized fractions and not in the dissolved fraction. The radionuclides are sorbed onto the surfaces of clay and zeolite particles. Because of the unique 240/239 signatures of the Pu isotopes, it was possible to identify the specific source (underground test sites) of the radionuclides. The cited evidence supports the proposal that transport has occurred over distances of at least 1,300 meters during the past 28 years. In the absence of an alternative interpretation or additional data, this work provides a clear example of rather rapid transport of radionuclides as colloids in volcanic rocks similar to those at Yucca Mountain. Perhaps of even more importance than the observation of colloid transport in volcanic rocks, the Panel was impressed by the possibility of testing transport models at the underground test sites of the Nevada Test Site (NTS) in both saturated and unsaturated volcanic units. As discussed in other parts of this report, such tests are essential to developing useful models for the TSPA and determining the associated uncertainties by comparison to natural systems.

On the basis of its review, the Panel recommends:

- The conduct of a careful analysis of the data of Kersting and Thompson (1997) to determine their applicability to the Yucca Mountain TSPA.

- The use of the data available at other sites at the NTS to perform tests of models used to describe radionuclide transport in the volcanic rocks of the site.

The Panel notes that the Project team has clearly identified colloid transport of actinides as an important issue (presentation by S. Brocum to the NWTRB in October of 1997). Evidently, a substantial amount of work has been completed, but the LANL report which will summarize the occurrences and effects of radionuclide migration via colloids is not scheduled for completion until October of 1998. The proposed work for transport and PA modeling (FY 1998 and beyond) will not be available for the TSPA-VA.

F. Disruptive Events, Criticality, and Climate Change

Disruptive Events

The three principal "disruptive events" that the TSPA-VA project is analyzing are:

- earthquakes;

- volcanism; and

- human intrusion.

Earthquakes

The effects of earthquakes at Yucca Mountain include, in principle, a wide range of phenomena depending on how large the postulated earthquake might be, when it might occur, whether ground shaking/acceleration or ground displacement (or both) might be important, and whether the effects are limited to disruption of the integrity of the waste in its canister or also includes effects during UZ transport or SZ transport of radioactive materials.

An extensive probabilistic seismic hazard analysis (PSHA) has been undertaken to understand the issue of how large the earthquakes might be at Yucca Mountain, when they might occur, and the characteristics of their effects. This PSHA is still in its final stages and will not be available for a few months; the Panel looks forward to reviewing it at that time.

In the meantime, the TSPA-VA team, using preliminary insights from earlier PSHA-type evaluations, has chosen to narrow their analytical effort to study principally only one key issue: the direct effect that a postulated earthquake might have on in-drift rockfalls that could impact an otherwise intact or nearly-intact canister and its contents. Enhanced waste degradation and enhanced mobility of the waste are the undesired endpoints being studied. The analysts will examine whether earthquake-caused rockfalls could make an important contribution in addition to effects in the non-seismic base-case scenario. Issues to be studied include damage to the waste package as a function of rockfall size (which can have larger effects at later times when the waste canister has lost significant integrity), and possible changes in seepage patterns into the drift.

The approach for the TSPA-VA is to perform an exploratory bounding-type analysis, to ascertain whether the effects are important enough to merit significantly deeper study.

Various indirect effects due to earthquake motion, such as changes in groundwater flow and transport patterns in either the unsaturated or saturated zones, will not be studied in detail in this TSPA-VA round. In part, this is due to the fact that the PSHA is not yet available and time is limited.

The Panel recognizes that this effort is still in an early stage, and looks forward to reviewing the work as it progresses. In particular, we expect to review both the direct-effect studies to determine if they require supplementing with more work later, and the indirect-effect issue to ascertain whether it can truly be dismissed.

Volcanism

The Basin and Range Province of the western United States is an active tectonic and volcanic region, and, indeed, there has been volcanic activity not very far from the proposed repository site at Yucca Mountain in quite recent times: within the past few thousand years. To understand both the frequency and the sizes/effects of potential volcanic activities of different types, the Project team commissioned the previously cited PVHA that enlisted the participation of most of the recognized experts in the field who could contribute to understanding the issues for the proposed repository (CRWMS M&O 1996c).

The Panel has studied this PVHA, which is well documented. Since none of the Panel members is an expert on volcanic hazards, there is no basis for the Panel providing a formal peer review of that work. The results of the PVHA suggest that volcanic activity that might affect the repository is quite unlikely; the aggregated results are that return frequencies are in the range of 10^{-7} to 10^{-9} per year, or even smaller, for the intersection of a volcanic event with the repository footprint. While the various experts have different models, and while several different types of volcanism could affect the repository, these PVHA results suggest that volcanism is very unlikely to be an issue for the repository.

Nevertheless, despite this quite low frequency, the TSPA project has undertaken an extensive effort to understand the effects of various volcanic scenarios on the repository. Much of this work was done, or well underway, before the results of the PVHA were available, and the work represents a substantial effort that has covered a large number of issues.

The work is in three parts. First, an exhaustive effort has been made to identify all of the possibly relevant scenarios, using a decision-tree-type or event-tree-type structure to differentiate among the scenarios. This has provided the basis for the second stage, which has been to identify a few scenarios for further analysis, basing the selection on criteria such as being reasonably comprehensive, conservative, and yet with enough breadth of coverage to assure that no key issues remain uncovered. Finally, the consequences of each of the scenarios selected for further analysis are to be analyzed (this stage is still underway, with the results not expected for a few months.)

The Panel's effort so far has been: (1) to review the logic of the approach, which seems reasonable; (2) to review the choice of scenarios for analysis, which choice seems sensible although it has not been possible to review that choice in detail because the full documentation is not yet available; and (3) to discuss the volcanism issues with the analysis team, so as to understand what is being attempted and why.

The analysis plan is ambitious, covering both potential direct effects of volcanic activity that might directly impact the waste in the repository, and indirect effects such as modifications to the geologic and hydrologic setting. A large amount of detail has been included in the models developed to date, and the work planned for the next few months will exploit this work-to-date to determine some reasonably good estimates or bounds on the potential consequences of several volcanic scenarios.

The Panel is looking forward to a review of the volcanism work when it is complete. As explained to the Panel, the TSPA team is attempting at this stage to do an analysis that will be sufficiently comprehensive to demonstrate with high confidence that volcanism is not important for the repository's overall performance. The TSPA team believes that the modeling work already accomplished, and the plans for the next few months, will provide such a demonstration.

Inadvertent Human Intrusion

The approach that the TSPA project will ultimately take in analyzing inadvertent human intrusion into the repository is still in limbo. The analytical approach applied in the License Application will depend on regulatory decisions by the EPA and the USNRC that have not yet been made. Specifically, until the EPA standard and USNRC's regulatory approach to implementing it are promulgated, the Yucca Mountain Project team will not know which human intrusion scenarios to analyze, which regulatory figures-of-merit to use, or the details of any other specific regulatory guidance. The need for regulatory guidance in this area is clear; because there is no way to predict human behavior in the distant future, no analysis can be "realistic" in either selecting its intrusion scenario(s) or assigning them probabilities -- thus the need for regulatory guidance.

Given the uncertainty in what the regulatory bodies will ultimately adopt, the approach that the TSPA-VA team is taking at this time seems eminently sensible. The project is temporarily assuming that the guidance in the report “Technical Bases for Yucca Mountain Standards” (National Research Council, 1995) will become the EPA/USNRC regulatory guidance.

That guidance suggests that the project not be required by regulation to analyze for human intrusion in a full probabilistic sense, because the probability per year of intrusion cannot be known. Instead, the suggestion is that the project be required to analyze the effects on overall repository performance from a single exploratory borehole (or perhaps a very small number -- two or three -- if that small number creates a scenario qualitatively different from the single-borehole scenario). The idea is to determine if such a modest campaign of exploration sometime in the distant future could compromise the performance that the repository would otherwise exhibit in terms of containment.

The guidance further suggests that only inadvertent future human intrusion be considered; that current-day exploration technology be assumed; and that the analysts assume that the exploration team somehow does not detect what it has encountered until the operation is complete. Then the explorers become suspicious and stop their campaign, but do not repair any damage to the repository underground. The analysts should ignore the effects of the intrusion on the exploration team themselves or their immediate environment (for example, from exposure to radioactive cuttings brought to the surface, either direct exposure or exposure due to subsequent dispersion), because such effects cannot differentiate between an excellent repository site/design and a poor one.

Because no regulatory guidance now exists, and because once that guidance is promulgated a full suite of analyses will become necessary, the TSPA-VA team's approach at this stage is to do some exploratory analysis, that is believed to be conservative and simplified. The approach, as described to the Panel, is that the analysts assume that a single exploratory borehole is drilled using typical modern drilling technology, that would pass from the surface directly through a waste package, extend all the way down to the saturated zone, and deposit radioactive waste at the bottom of the borehole directly at the top of the SZ. This waste would then be available to migrate in the SZ and toward the accessible environment. The question will then be asked as to whether such a scenario, that is assumed conservatively to bypass the unsaturated zone entirely, produces important additional radionuclide transport to the accessible environment when compared to the no-human-intrusion base case. The time in the future when such an exploratory hole is assumed to occur will be varied, to assess which future time period might be "worst" in terms of consequences.

While the Panel has not had the opportunity to review the details of this analysis, because it is still underway, the approach makes eminent sense. Insights gained from this preliminary analysis can indicate whether a much more detailed analysis of human intrusion scenarios will be needed, assuming that EPA and USNRC adopt the regulatory approach suggested by the Committee on Technical Bases for Yucca Mountain Standards (National Research Council, 1995). The Panel will await the opportunity, over the next few months, to review the details as the analysis proceeds.

Criticality

The TSPA-VA constitutes the first attempt to address the issue of criticality at Yucca Mountain through performance assessment; it was not addressed systematically in TSPA-95 or earlier TSPAs. The TSPA-VA team will not attempt to integrate the criticality analyses with the larger PA model, but instead will perform a set of side analyses of criticality scenarios as a sensitivity study in parallel with the mainline analysis of future repository performance. That is, criticality scenarios will not be incorporated into the mainline models for TSPA-VA, but will be analyzed separately.

In brief, the criticality problem is that a very large number of critical masses, of either plutonium or uranium-235, will be emplaced in the waste canisters, and many other critical masses of various fissile nuclides will grow into the waste over the eons through radioactive decay of parent nuclides. Although the material as originally emplaced will be in configurations that will be designed to preclude criticality, it is necessary to determine whether a critical mass could be reassembled later in time after the engineered barrier features degrade.

As the Panel pointed out in its first report, the task of TSPA-VA Project team in this area should be some combination of the following: (1) to perform a set of realistic analyses of all of the various potential criticality scenarios, or (2) to analyze only a subset of the potential scenarios and then to argue that this subset bounds the larger set of scenarios that are not analyzed; or, where appropriate, (3) to produce bounding analyses of some scenarios if such would be adequate for the purposes of the overall TSPA-VA project.

The Project team has approached this difficult analysis task in four steps. First, the Project team has identified three physically distinct regions where criticality might occur in the far future: in-package criticality (after degradation of the packages or of their contents); near-field, in-the-drift criticality after material might migrate out of the canisters into the drift space; and far-field criticality, defined as anywhere outside the drift. Secondly, the team has differentiated in a complex decision-tree or event-tree format the full range of potential scenarios, in each of the three regions, that might occur given different postulated future events and processes. Using this complex event-tree structure, the third step has been to choose a small number of potential scenarios for analysis during this round (TSPA-VA). The fourth step, now underway, will be to analyze each of these scenarios in a realistic manner, but using conservative assumptions where appropriate.

It is important to describe the two key explicit assumptions with which the TSPA-VA team is operating that: (1) that it can later be shown that the few analyzed scenarios truly do "bound" all of the others, in the sense that the doses/risks from them exceed the doses/risks from all of the others; and (2) that none of the scenarios analyzed will contribute importantly to the overall doses/risks from the proposed repository when compared to the no-criticality base-case analysis.

If both of these assumptions are correct, the issue of criticality will have been shown to be "unimportant," at least in a regulatory-compliance sense.

In principle, any specific criticality scenario can be screened out if either its likelihood is found to be exceedingly small, or its dose/risk consequences are found to be minor compared to the base-case behavior of the repository, absent that scenario. As the Panel understands the TSPA-VA team's approach, this logic will be used to eliminate many, if not all, scenarios, thereby enabling the analysts to dispense with criticality concerns for the repository. (Of course, care must be taken that one does not screen out a myriad of small scenarios one-by-one while overlooking the possibility that they will add up to an important impact; the likelihood for error inherent in such a "divide-and-conquer" approach is an ever-present danger when choosing to examine only a few scenarios among a much larger set.)

Progress to date has been significant. The TSPA-VA team has completed developing the set of scenarios and has selected a subset for analysis in this round. The team has recently published an account of its work (CRWMS M&O 1997c) and is now embarking on the analysis itself, which will be designed to estimate both the likelihood and the dose/risk consequences of each chosen scenario. The TSPA-VA team has selected for analysis six different in-canister scenarios and one each in the near-field and far-field regions.

Over the past year, in the course of differentiating among the scenarios and choosing the few to analyze, the TSPA-VA team has reached some important conclusions about the various phenomena. They now believe that if any criticality scenarios turn out to be important, they will be the in-canister ones; they believe that it will be possible to show, in this TSPA-VA round, that all scenarios in both the near field and the far field can be dismissed on the basis of either probability or consequences, and perhaps both. In particular, criticality scenarios in the near field (in-the-drift after material migrates out of the canisters into the drift space) seem so far likely to produce only very minor increases in consequences over the no-criticality base-case scenario. Further, these scenarios have at most a rather small likelihood of occurring -- although these likelihoods are difficult to estimate, especially the likelihoods that neutron-absorbing materials might be separated from the fissile materials enough to produce the criticality scenario(s). Similarly, the far-field scenarios appear to be of concern, if at all, only for time periods beyond a million years, because the important processes that might segregate and/or reconcentrate a critical mass and eliminate any neutron-absorbing materials in the far field appear to be very slow, taking place in the millions-of-years range. (These conclusions, if supported by further work now under way, will require careful review by the Panel.)

The in-canister scenarios remain as the most likely concern. Here, the TSPA-VA team is developing details of how canister-failure mechanisms might introduce moderator (water), displace the neutron-absorbing material, assemble the fissile material into a critical configuration, and sustain all of this to produce a fissioning system. In the opinion of the Panel it is unlikely that anybody will ever be able to "predict" the details of how canister failure and the other phenomena might occur, and to assign split-fraction probabilities to the various failure scenarios and the subsequent events. Even though there is a sound scientific understanding of the key phenomena, such as differential chemical-separation effects as a function of conditions (pH, Eh, temperature, etc.) and critical-assembly behavior, it is more likely that the analysis team will be successful because the TSPA-VA team will be able to show, with confidence, that the bounds it can place on consequences and/or probabilities, taken together, are acceptably minor. If not, and the specific details need to be understood, the situation could be beyond the capabilities of current knowledge, especially insofar as it would be necessary to understand the details various future canister-failure scenarios.

The Panel expects to review the details of the criticality work over the next few months, as the Project team completes its analysis of the various scenarios. We will try to be especially attentive to whether the scenarios chosen are a reasonable set; whether any conservative or bounding-type assumptions are well chosen and used properly; and whether the mix of consequence-type arguments and likelihood-type arguments holds together coherently.

To summarize our comments about the criticality work to date, the Panel believes that the two key elements of the approach above -- allowing criticality to be studied through side analyses instead of in the mainline TSPA modeling, and developing a few scenarios for analysis in order to bound the problem -- are both sensible. The project should be commended for the logic adopted in the work being undertaken.

Regulating Against Criticality

Another important issue concerns the relevant standard to be used in evaluating the risks associated with criticality. In its first review report, the Panel observed that the USNRC regulations adopted many years ago for evaluating the possibility of criticality in deep-geological repositories such as that proposed at Yucca Mountain, imply that it is necessary to preclude criticality with high confidence. Unfortunately, in our view, the regulations, as written, do not clearly indicate whether they were intended to apply to the operational phase (pre-closure), the post-closure phase, or both. The Panel urged that the Project team request that the U.S. NCR staff clarify this situation.

During the intervening months, much progress has been made on this issue. The Panel is gratified and will monitor the evolution of the situation over the next year. Our reason for assigning this topic high importance is that, as the Panel stated in its first report, we believe that, depending on the figure-of-merit used in the regulations for the proposed Yucca Mountain repository, it may be determined whether the proposed repository "passes" or "fails" depending on the specific details to a much greater extent than for any other of the important phenomena that may occur in the future. Specifically, if the regulations require that the repository design "preclude" criticality from occurring within Yucca Mountain for all future times, or for any regulatory time period beyond when canister failure begins, the Panel believes that it may be impossible to demonstrate whether the facility complies.

As stated in our first report, the Panel's judgment on the above is based on the following (preliminary) observation. Despite all of the best efforts that the criticality modelers will bring to bear on the subject, it is our judgment that it likely will not be possible to preclude criticality processes with high confidence over the full future time covered in the TSPA. This is likely the case even if only a 10,000-year regulatory period is to be covered, and all the more true if much longer times, such as a million-year period, require study. This is because the specific details of the ways that the canisters may fail, and the ways that materials may chemically interact and move (both in-canister and in-drift), may not be knowable in enough detail.

Climate Change

The TSPA-VA Project team has not completed sufficient work on how climate change might affect the long term behavior of the proposed Yucca Mountain repository to provide revisable material for the Panel. Therefore, our review of this topic is deferred.

G. Biosphere, Doses, and Health Risks

Since issuance of its initial report, the Panel has been provided with the following reports that contain details on progress in the development of the biosphere components of the TSPA-VA:

- *Total System Performance Assessment - Viability Assessment (TSPA-VA), Methods and Assumptions* (TRW, 1997a); and

- *Biosphere Abstraction/Testing Workshop Results* (TRW, 1997b).

The Panel was also provided a transcript of the meeting of the Panel on Environmental Regulations and Quality Assurance, Nuclear Waste Technical Review Board, that was held on October 21, 1997.

On the basis of our reviews of these reports and related documents, the Panel offers the following comments and recommendations related to the methods and procedures that will be used in assessing the doses/risks to the public.

Assessing Doses and Health Risks

In the case of performance assessments for the proposed Yucca Mountain repository, it is possible that the EPA and the USNRC will provide the TSPA-VA team with specific values for the dose conversion factors and risk coefficients that are to be used.
Even so, the DOE and the TSPA team should seek to develop realistic estimates, with the objective of reaching an understanding of the conservatisms that underlie, and have been incorporated into, the dose conversion factors for each of the critical radionuclides as well as the coefficients for converting these dose estimates into the related risk. At the same time, the Panel wants to make it clear that it is not seeking to imply that the TSPA-VA team should develop new more realistic dose conversion factors and risk coefficients; rather it is to encourage the TSPA-VA team to be aware of the related conservatisms, to quantify them at least in a cursory sense, and to be prepared to discuss and evaluate their implications in terms of the outcome of the TSPA-VA.

Difficulty of the Task

The next comment pertains to the difficulties anticipated by the M&O staff in estimating the doses to population groups who may be exposed. In Section 1.1.2 of the Workshop report (CRWMS M&O, 1997b), the statement is made that:

> In the TSPA computational code it was a simple calculation to convert concentration of each radionuclide in the groundwater to dose. The dose for each radioisotope could be readily generated by simply taking the product of the dose conversion factor (DCF), the concentration of that radionuclide in the groundwater and the quantity of drinking water. The total dose was arrived at by summing this product over all radionuclides.

The Panel does not agree that this process is as “simple” as implied. As discussed below, unless care is exercised many of the errors, uncertainties, and conservatisms associated with making such estimates may not be recognized. Additional conservatisms and uncertainties will be introduced, as noted above, in converting the dose estimates into risk estimates.

Degree of Conservatism Being Sought

Closely associated with these topic is the degree of conservatism that is being sought in developing the dose/risk estimates. Although most of the analyses in the TSPA-VA appear to be directed to the development of "best estimates," Section 1.3 of the Workshop report (CRWMS M&O, 1997b) indicates that:

> Approximations and systematic errors (in the Biosphere 'add-in' model) have to be shown to provide predictions of dose that will be conservative.

Although the Panel agrees that conservatisms need to be incorporated into the standards or regulations, we do not agree that they should be incorporated into the dose/risk assessments. In fact, every effort should be made to make these assessments as realistic as possible. This was one of the points made by Dr. Marsha Sheppard of the Atomic Energy of Canada Whiteshell Laboratories during the Workshop cited above. As noted earlier in this report of the Panel, this was also one of the implications of the wording in the original EPA Standards, 40 CFR 191.13 (a), as cited in Section II (U.S. EPA, 1985). Although now remanded, these Standards clearly stated that "unequivocal proof of compliance is neither expected nor required because of the substantial uncertainties inherent in such long-term projections. Instead, the appropriate test is a reasonable expectation of compliance based upon practically obtainable information and analysis." The regulations of the USNRC (1983) followed a similar pattern in stating that "While these performance objectives and criteria are generally stated in unqualified terms, it is not expected that complete assurance that they will be met can be presented. A reasonable assurance, on the basis of the record before the commission, that the objectives and criteria will be met is the general standard that is required." Neither the EPA standards nor the supporting USNRC regulations imply that the risk/dose assessments should be calculated on a conservative basis.

Magnitudes of Conservatisms and Associated Uncertainties

It is important the TSPA team recognize the magnitudes of the conservatisms that have been incorporated into the existing dose conversion factors and risk coefficients. In this regard, the BEIR-V Committee (National Research Council, 1990) has cautioned that the

> … methodology and values given by International commission on Radiological Protection (ICRP) (for calculating the doses due to the internal deposition of radionuclides) were assembled for radiological protection purposes. Thus, the values chosen for the various parameters are conservative; that is, they can lead to overestimates of risk factors. These values may not be appropriate for estimation of risk when the organ and tissue doses received by exposed individuals are considered. (pages 40-41).

Similar words of caution have been expressed by the Committee on an Assessment of CDC Radiation Studies (National Research Council, 1995), when they stated that

> The largest dose will be to organs that accumulate and retain the radionuclide. However, the variability in absorption of the ingested radionuclides in the gastrointestinal tract is responsible for the greatest uncertainty in the potential dose. Because radiation guidelines are usually conservative, it is likely that the commonly used absorption factors overestimate the amount of the radionuclide that is absorbed and hence the organ dose. (page 43).

The Committee on an Assessment of CDC Radiation Studies also recommended that: (1) "In assessing exposure and absorbed dose, uncertainty should be expressed for physical, biological, and computational methods. The calculations of uncertainty should be propagated throughout all calculations…"; (2) "In obtaining measures of propagated errors, procedures for incorporating methods of assessment of uncertainty for physical and biologic results are required." (page 49); and (3) risk assessors should recognize that "Traditionally, radiation protection guidelines are predicated on a linear dose response, which assumes that the harmful effects of radiation are linearly related to the dose and that there is no threshold dose. Most experts believe this assumption is conservative; that is, it overestimates the effects of ionizing radiation at low doses because it ignores the potentially beneficial effects of the body's repair mechanisms." (page 43).

Still another conservatism is that resulting from the use of the committed dose concept, particularly for radionuclides with long effective half-lives, as is the case with ^{237}Np and ^{239}Pu. According to the National Council on Radiation Protection and Measurements (NCRP) (1993, page 25), the use of this concept "will overestimate by a factor of approximately two, or more, the lifetime equivalent dose or effective dose."

Adding support to these concerns is the recent action by the National Radiological Protection Board, United Kingdom, to develop an independent set of RBE values for use in risk assessments involving exposures from neutrons, as contrasted to applying those that have been developed for purposes of radiation protection (Edwards, 1997).

Acceptability of Health Endpoint

At this stage, it is anticipated that the standards being developed for the proposed repository will be expressed in terms of dose and/or risk limits that are based on the probability of fatal cancers as the health endpoint. Although this was the endpoint commonly used in the past (ICRP, 1977), newer recommendations of organizations such as the NCRP and the ICRP are based on what is called the "total detriment." This includes considerations of both morbidity and mortality, as well as years of life lost (ICRP, 1991).

If fatal cancers are considered to be a surrogate for other health endpoints, the basis for this selection needs to be explained. The issue of what endpoints should be considered, including fatal and nonfatal cancers and other late effects of ionizing radiation, are appropriate topics for discussion between the project staff and the regulators, and should be considered by the regulatory agencies as issues to be raised in the public input processes associated with development of the standard. To the extent that considerations of this type may impact on the acceptability of the TSPA, the Panel encourages the TSPA team to keep these factors in mind and to be prepared to address them.

Identification of Significant Radionuclides

The conservatisms cited above, coupled with other considerations, have led the Panel to question whether the TSPA-VA Team has devoted sufficient effort to the identification of those radionuclides that are most important in assessing the potential impacts of the proposed repository. The current list needs to be shortened and the key radionuclides need to be identified. Included in this process should be a thorough discussion of the scientific basis for each such selection. One radionuclide that serves as a source for these comments is ^{129}I.

According to the NCRP (1985, page 41), "The low specific activity (0.17 μCi/mg) of ^{129}I and the restricted capacity of the normal human thyroid to store iodine, limit the hazard from ^{129}I." Based on these observations and studies of the effects of ^{129}I in animals, the NCRP concluded that "^{129}I does not pose a meaningful threat of thyroid carcinogenesis in people."

For these reasons, the Panel believes that, while ^{129}I will still have to be considered by the TSPA-VA team and appropriate dose estimates made, the team should be aware of the views of the NCRP. Similar reviews should be conducted of the detailed physical, biological, and chemical information on each of the other 39 radionuclides currently on the list of those considered important by the TSPA-VA Team. These types of issues should be analyzed and discussed with the regulators to ensure that there is a scientifically sound basis behind whatever regulations are adopted. The goal should be to define a sound scientific basis for the selection of each radionuclide considered to be important.

Relative Importance of Dose/Risk Uncertainties and Conservatisms

In summary, the Panel believes that it is important for the TSPA team to recognize that the conservatisms enumerated above and to document and quantify the associated uncertainties. Although predictions of future climatic conditions and geologic developments, and the anticipated behavior of population groups, are important, the biosphere dose/risk issues appear to the Panel to offer equal challenges. In certain cases, the magnitude of the uncertainties and potential errors in the pathway, dose and risk estimates may equal those involving assessments of the performance of the natural and engineered barriers .

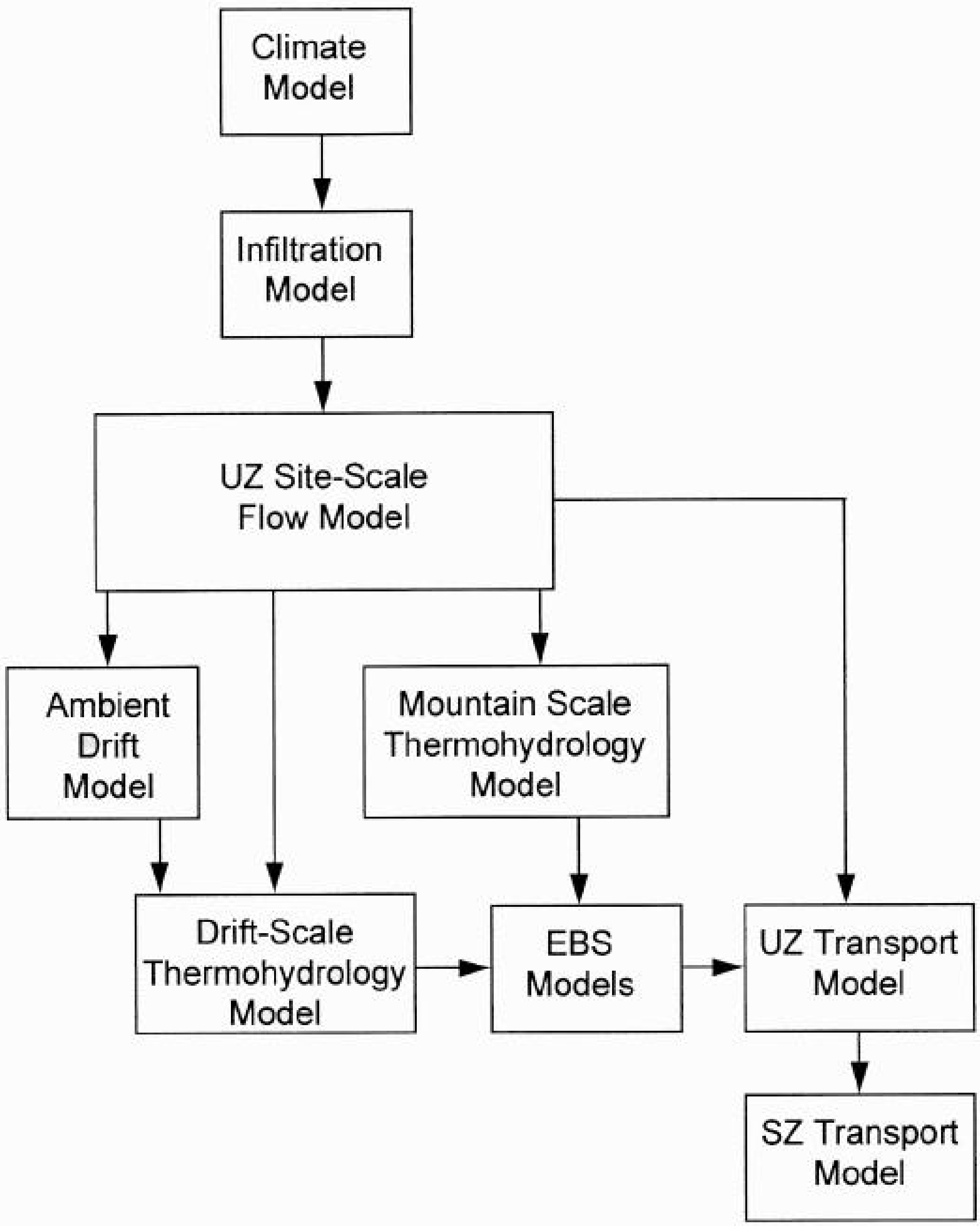

Figure III-1. Relationships between various computer models being used in the analysis of Yucca Mountain (from Bodvarsson et al., 1997b).

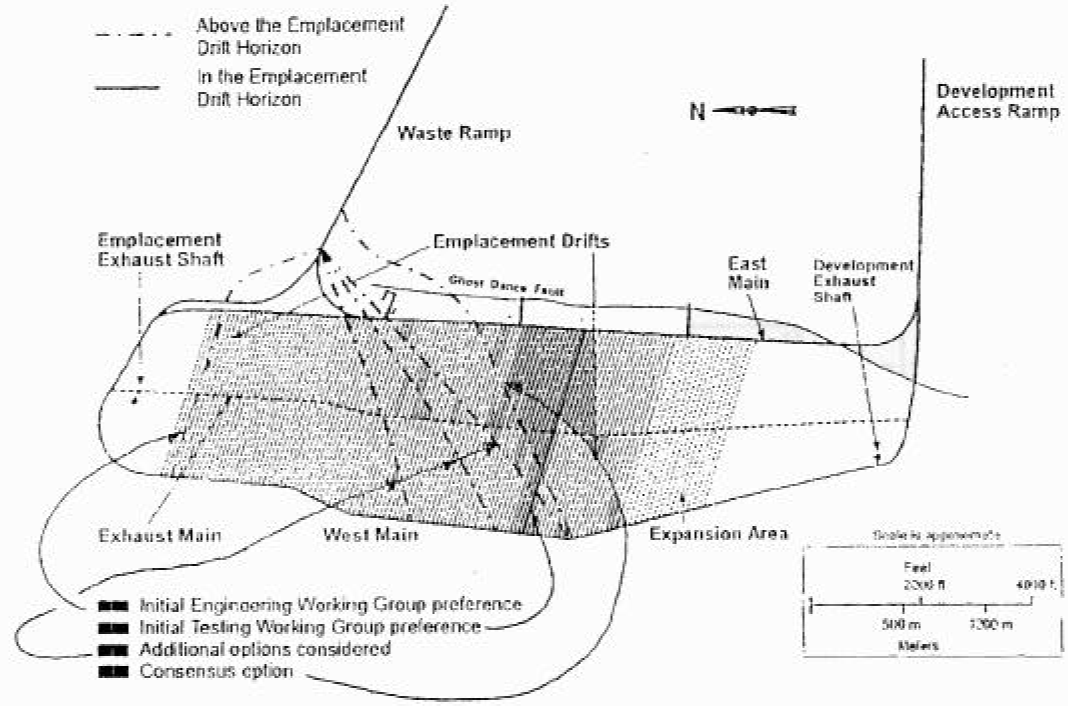

Figure III-2. Location of East-West drift selected from the enhanced characterization of the repository block (ECRB) showing various options considered in the analysis.

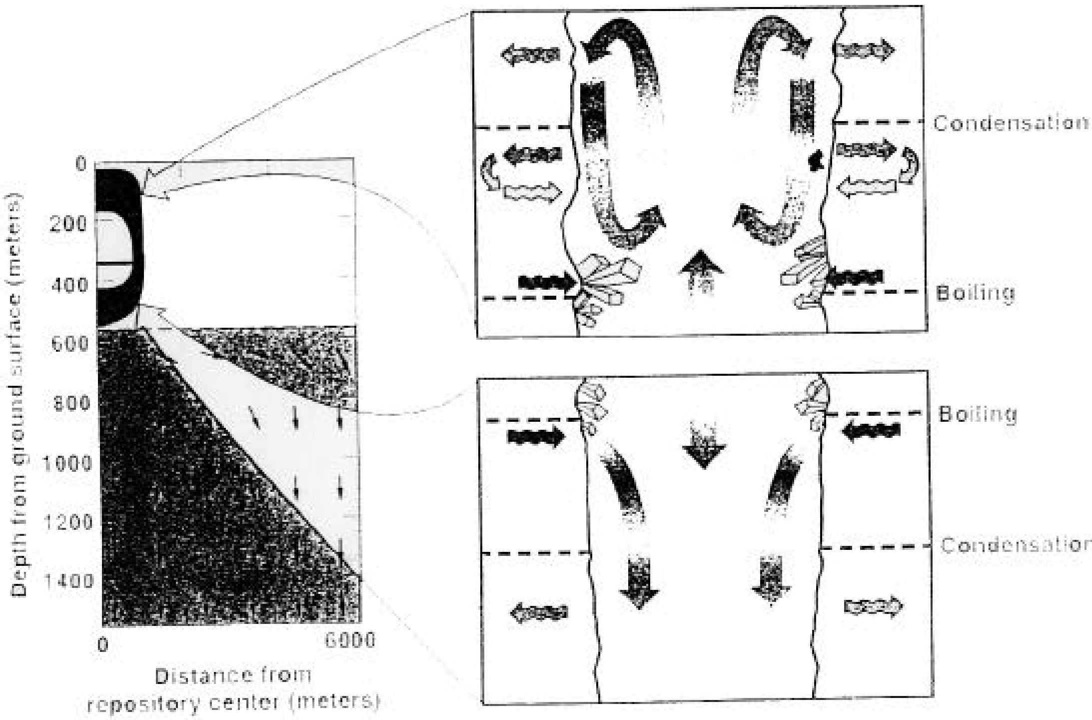

Figure III-3. Schematic drawing of heat pipes and geochemistry regimes at 1000 years post closure (from Glassely et al. 1997).

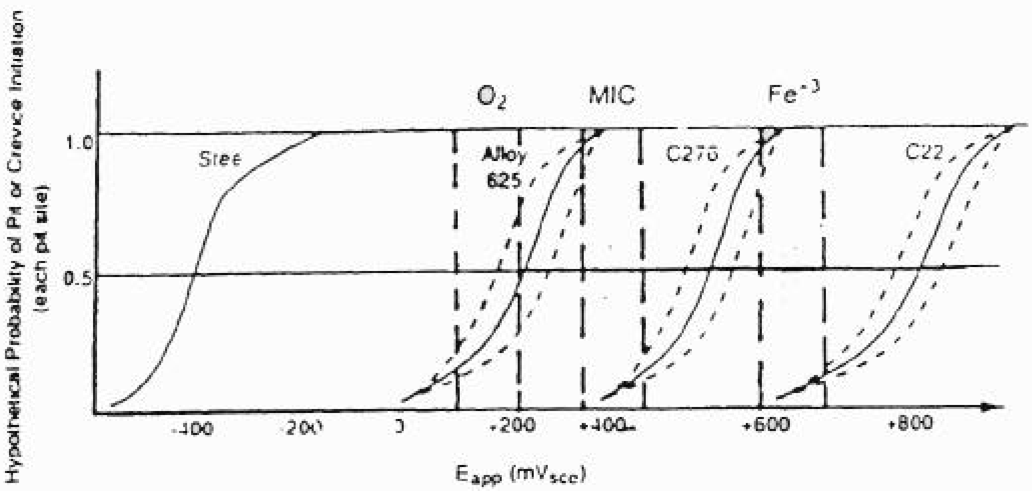

Figure III-4. Crevice corrosion in metals (from J.R. Scully, U.S. Nuclear Waste Technical Review Board Meeting, October 23, 1997).

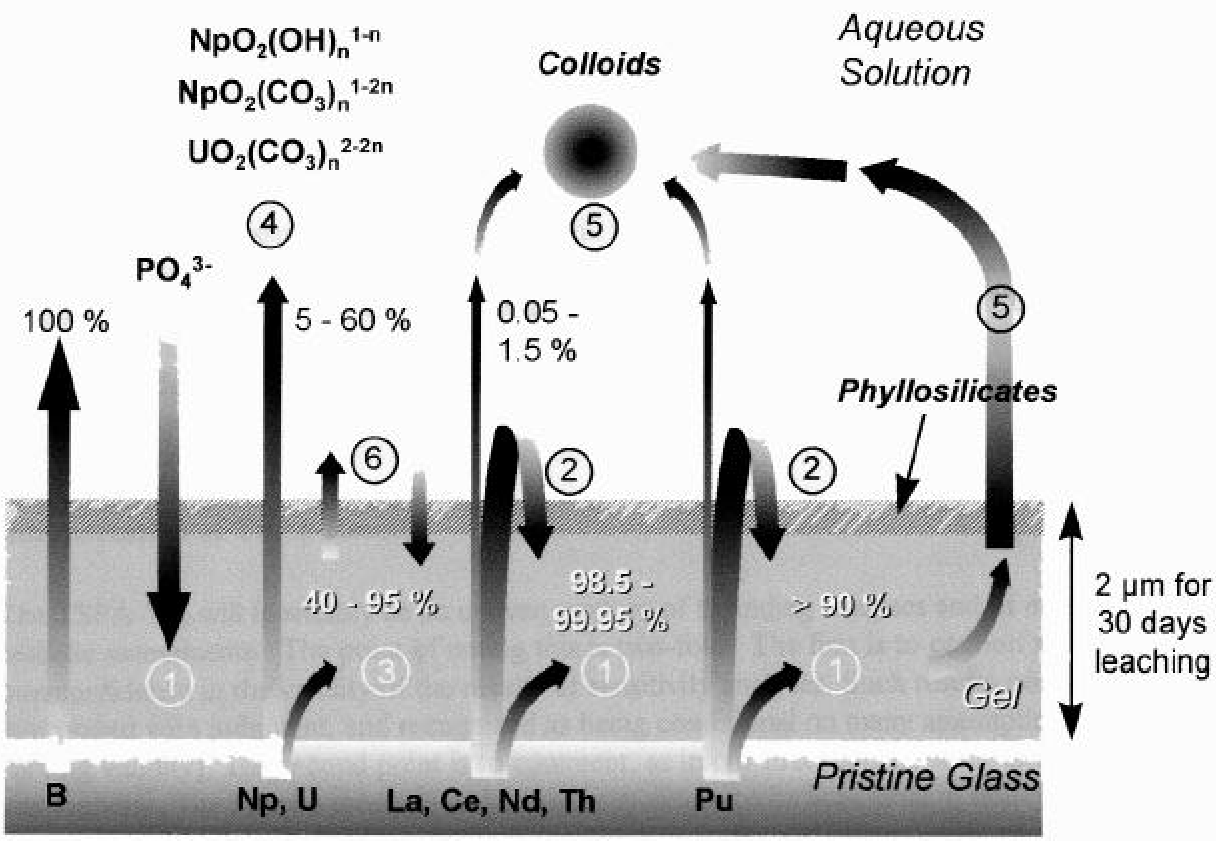

Figure III-5. Principal mechanisms involved in controlling the mobility of the lanthanides and actinides during the leaching of R7T7 nuclear glass under simulated geological disposal conditions: (1) Coprecipitation/Condensation; (2) Chemisorption; (3) Precipitation of phosphate or oxide/hydroxide phases; (4) Complexation; (5) Colloid transport; (6) Ion exchange. Figure courtesy of Thiery Advocat (CEA) (Menard et al. in press).